JUN 16 1987

1051-H-11

Y3.N88:25/4700/draft/v.3

NUREG/CR-4700
SAND86-1135
Volume 3
Printed May 1987

DRAFT REPORT FOR COMMENT

COMPLETED
561
ORIGINAL

Containment Event Analysis for Postulated Severe Accidents: Peach Bottom Atomic Power Station, Unit 2

C. N. Amos, J. M. Griesmeyer,
A. M. Kolaczkowski

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550
for the United States Department of Energy
under Contract DE-AC04-76DP00789

Prepared for
U.S. NUCLEAR REGULATORY COMMISSION

100

NOTICE

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082,
   Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Information Support Services, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

DRAFT FOR COMMENT

NUREG/CR-4700
SAND86-1135
Volume 3

CONTAINMENT EVENT ANALYSIS FOR POSTULATED SEVERE ACCIDENTS: PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

C. N. Amos*
J. M. Griesmeyer
A. M. Kolaczkowski**

Printed May 1987

Sandia National Laboratories
Albuquerque, NM 87185
Operated by
Sandia Corporation
for the
U. S. Department of Energy

Prepared for
Division of Reactor System Safety
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
Under Memorandum of Understanding DOE 40-550-75
NRC FIN A1322

---

*Technadyne Engineering Consultants, Inc., Albuq., NM
**Science Applications International Corporation, Albuq., NM

I

BLANK PAGE

REQUEST FOR COMMENT

This report, NUREG/CR-4700, "Containment Event Analysis for Postulated Severe Accidents," was prepared for the U.S. Nuclear Regulatory Commission by the Sandia National Laboratories and its subcontractors.

The methods and results set forth in this report are being used by the NRC to support the development of the Reactor Risk Reference Document (NUREG-1150) and will be used in areas of broad public interest such as probabilistic risk analyses, emergency response planning, siting, NRC safety goal applications, and cost/risk/benefit analyses--indeed, wherever risks to public health need to be considered in regulatory applications. Thus, it is considered imperative that an opportunity for public comment on the results as presented in the report be provided. Comments should be sent to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Joseph Murphy, Division of Reactor System Safety. These comments will be most useful to the staff if they are received by August 1, 1987.

Of particular interest to us is the receipt of comments on the methodology, and results, related to uncertainty analysis. One criticism voiced with respect to the Reactor Safety Study was its lack of an uncertainty analysis. We have included an uncertainty analysis, but we are sure that its nature will be the subject of lively debate. We welcome this, and solicit constructive advice and criticism.

The NRC hereby expresses its great appreciation to all participants in this study for their considerable efforts, as well as to all who will take the time and effort to provide it with comments on this report.

DFRoss

D. F. Ross, Deputy Director
Office of Nuclear Regulatory Research

ABSTRACT

A study has been performed as part of the Severe Accident Risk Reduction Program (SARRP) to investigate the response of a particular boiling water reactor with a Mark I containment (Peach Bottom Unit 2) to postulated severe accidents. A detailed containment event tree for the Peach Bottom plant has been developed to describe the various possible accident pathways that can lead to radioactive releases from containment. Data and analyses from a large number of NRC and industry-sponsored programs have been reviewed and used as a basis for quantifying the event tree, i.e., determining the likelihood of the pathways at each branch point for a variety of accident sequence initiators. A generalized containment event tree code, called EVNTRE, has been developed to facilitate the quantification. The uncertainty in the results has been examined by performing the quantification three times, using a different set of input each time to represent the variation of opinion in the reactor safety community. In the so-called "central" estimate, the likelihood of early containment failure (occurring before or within a short time after reactor vessel breach) was found to be significant because of the possible occurrence of the following phenomena that can threaten containment integrity: (1) meltthrough of the drywell shell caused by thermal attack from core debris, and (2) drywell overpressurization caused by rapid depressurization of the reactor vessel in combination with other events such as direct heating. However, uncertainties surrounding these issues could cause the early failure likelihood to be significantly lower than in the central estimate. This work supports NRC's assessment of severe accident risks to be published in NUREG-1150.

PREFACE

Because of the time constraints imposed on this work to meet the Nuclear Regulatory Commission's schedule for publication of NUREG-1150, this draft report has not yet received the full level of peer and management review customarily accorded to reports issued by Sandia National Laboratories. The review will be completed and corrections made if necessary prior to the final publication.

BLANK PAGE

TABLE OF CONTENTS

SECTION PAGE

1. INTRODUCTION.. 1

2. METHOD.. 4
   - 2.1 General Description and Observations....... 4
   - 2.2 Specific Example: Peach Bottom TBUX........ 8
   - 2.3 Summary of Plant Features and Accident Sequences................................. 8

3. RESULTS... 14
   - 3.1 Cut Set Groups TBUX and TBU................ 15
   - 3.2 Cut Set Group TB........................... 18
   - 3.3 Cut Set Group TCUX......................... 20
   - 3.4 Cut Set Group TCSX......................... 21
   - 3.5 Cut Set Group TCSRX2....................... 23
   - 3.6 Cut Set Group TCSRV23...................... 24

4. SUMMARY AND CONCLUSIONS.............................. 26

5. REFERENCES.. 50

App. A DETAILED RESULTS OF THE CONTAINMENT EVENT TREE ANALYSIS.. A.1-54

App. B PEACH BOTTOM CONTAINMENT EVENT TREE DESCRIPTION....................................... B.1-116

App. C SUPPORTING CALCULATIONS FOR THE SARRP PEACH BOTTOM EVENT TREE................................. C.1-348

App. D SAMPLE EVENT TREE PROBLEM........................ D.1-378

App. E EVNTRE INPUT FOR PEACH BOTTOM.................... E.1-395

LIST OF FIGURES

FIGURE PAGE

3.1 Dominant Accident Progression Bins, Short-Term Station Blackout........................ 46

3.2 Dominant Accident Progression Bins, Long-Term Station Blackout........................ 47

3.3 Dominant Accident Progression Bins, ATWS Followed by No Injection...................... 48

3.4 Dominant Accident Progression Bins, ATWS With Initial Injection....................... 49

LIST OF TABLES

TABLE		PAGE
2.1	Summary of CET Events for Peach Bottom........	28
2.2	Nomenclature Used in Plant Damage State Names.	32
2.3	Mean Frequency of Plant Damage States	33
3.1	Central Evaluation of Peach Bottom Containment Performance.......................	34
3.2	Optimistic Evaluation of Peach Bottom Containment Performance.......................	42
3.3	Pessimistic Evaluation of Peach Bottom Containment Performance.......................	44

LIST OF ACRONYMS AND ABBREVIATIONS

ADS	Automatic depressurization system
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated transient without scram
BWR	Boiling water reactor
CCI	Core-concrete interactions
CDB	Coolable debris bed
CET	Containment event tree
CF	Containment failure
CLWG	Containment Loads Working Group
CPWG	Containment Performance Working Group
CRD	Control-rod drive
ECCS	Emergency core cooling system
ESF	Engineered safety feature
HPCI	High pressure coolant injection
HPSWS	High pressure service water system
IDCOR	Industry Degraded Core Program
LLH	Limited Latin hypercube (uncertainty analysis method)
LOCA	Loss-of-coolant accident
MSIV	Main steam isolation valve
NPSH	Net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
OCP	Optimistic/central/pessimistic (method for uncertainty analysis)
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
QUEST	Quantitative Uncertainty Estimation of the Source Term Program
RCIC	Reactor core isolation cooling system
RHR	Residual heat removal
RPV	Reactor pressure vessel
RSS	Reactor Safety Study
RSSMAP	Reactor Safety Study Methodology Applications Program
SARRP	Severe Accident Risk Reduction Program
SASA	Severe Accident Sequence Analysis Program
SAUNA	Severe Accident Uncertainty Analysis Program
SCG	Senior consultant group
SLCS	Standby liquid control system
SRV	Safety/relief valve
STCP	Source Term Code Package
USI	Unresolved safety issue

1. INTRODUCTION

This report documents analyses performed under the Severe Accident Risk Reduction Program (SARRP) to investigate the response of containments at commercial nuclear power plants to severe accidents (beyond the design basis). These analyses focus on identifying the various pathways that could lead to the release of fission products beyond the containment boundary and on estimating their frequencies of occurrence. The analyses are part of an up-to-date characterization of nuclear plant risk [1] that will serve as a principal input to a forthcoming report by the U.S. Nuclear Regulatory Commission (NRC) Staff, NUREG-1150 [2]. It is anticipated that NUREG-1150 will provide a technical basis upon which to consider future decisions regarding changes to regulations and regulatory practices.

The analyses described in this report represent an updated and broader perspective on the significance of the fission product source terms estimated in NUREG-0956 [3], BMI-2104 [4], and in subsequent analyses performed for SARRP [5] by examining the ways in which containment failures leading to source terms of different magnitudes can occur.

One might ask, "Why is there a need for such an updated perspective on the various accident pathways that are possible?" To answer this question, we must first consider the fact that a limited number of analyses were performed as part of the BMI-2104 work. The analyses were selected on the basis of previous knowledge of possible large risk contributors and included a transient with loss of long-term heat removal (TW sequence), a large loss-of-coolant accident (LOCA) with loss of injection (AE sequence), and an anticipated transient without scram (ATWS) scenario. However, plants such as Peach Bottom have recently made a number of both design and operational changes. In addition, probabilistic risk assessment (PRA) and containment analysis methodology and investigations have further evolved providing newer insights. These two factors, in combination, have been reflected in the accident sequence and containment analysis update as part of the NUREG-1150 work. The resulting insights show that instead of TW or AE type sequences, variations of a total loss of all ac power (station blackout) and to a lesser extent, ATWS, dominate the core damage frequency and risk for Peach Bottom. With a change in focus to a number of station blackout scenarios, an update of the most important accident pathways is necessary. Questions regarding containment venting and late recovery of ac power are examples of analysis areas requiring investigation for their effect on containment failure, release pathways, and ultimately the magnitude of the source term.

These questions raise the need for a systematic identification of the various pathways that the accident can take and assessment of the likelihood, or probability, of each. Such an analysis provides a basis for evaluating whether the source terms developed for a particular fission product release pathway cover the range of risk-significant source terms that could result from that accident sequence.

The overall objectives of this study are (1) to identify accident progression pathways (i.e., combinations of accident sequences and containment events) that delineate source terms that may be important to risk, (2) to estimate the frequencies of those pathways and hence the frequencies of the resulting source terms, (3) to identify accident pathways for which additional source term calculations are needed, and (4) to provide direct input to the SARRP evaluation of severe accident risks [1]. At the current time, these objectives have been implemented through assessment of four reference plants: Surry (PWR subatmospheric containment), Peach Bottom (BWR Mark I), Sequoyah (PWR ice condenser), and Grand Gulf (BWR Mark III). This volume describes the analyses for the Peach Bottom plant. Each of the other plants is treated in a separate volume of this report.

With regard to the nature of the sequences considered, two aspects should be noted. In the current analysis, accidents initiated by external events (e.g., earthquakes, fires, flooding, etc.) are not explicitly treated since no analyses of the frequencies of these events or of the capabilities of plant systems and structures to withstand them have been performed. It is planned for external events to be included in the final version of this report. In addition, an in-depth evaluation of operator actions that might be taken to respond to core melt accidents to mitigate their consequences has not been performed. Plants such as Peach Bottom have procedures that provide only a limited set of instructions to the operators past the onset of melting of the core; it is beyond the scope of the present analysis to devise more detailed procedures and to evaluate their effects.

In Section 2, the method used to achieve the objectives stated above is described. First a general description of the containment event tree is provided that also addresses how the branches are quantified. Some special considerations and observations are also discussed as well as the interface between the updated accident sequence analyses and this work. In Section 3, the results are presented in terms of frequencies of containment failure modes and accident pathways for the various accidents that were assessed to be potential contributors to risk for Peach Bottom. In Section 4 the results are summarized along with a discussion of those that are particularly significant. Detailed results are provided in Appendix A.

Appendix B provides the detailed description of the Peach Bottom containment event tree. This includes the justification for the evaluation of the probability of each event considered in the tree. A sample problem to facilitate the explanation of the manner in which the input for the event tree is structured is given in Appendix D. Appendix E includes the actual computer code input that forms the Peach Bottom containment event tree.

The containment event trees developed by SARRP have been reviewed by a committee commissioned by NRC and consisting of representatives from the University of Wisconsin, Battelle Columbus Laboratories, Oak Ridge National Laboratory, Brookhaven National Laboratory, and Pickard, Lowe & Garrick, Inc. The review group investigated the reasonableness of the event trees (scrutability, completeness, consistency, and level of detail) and the reasonableness of the quantification (traceability, documentation completeness, coverage of sources, and consistency of approach). The review uncovered a few errors of logic and omission that have subsequently been corrected. Their detailed comments are provided in References 6 and 7. The containment event tree was also reviewed by representatives from Philadelphia Electric Company, whose comments are available in Reference 8.

2. METHOD

2.1 General Description and Observations

In traditional PRAs, the accident pathways that contribute to risk are described by two types of event trees. "System event trees" are used to define the spectrum of accident sequences (i.e., the combinations of accident initiators and subsequent system failures) that can lead to core melting. "Containment event trees" are used to define the containment failure modes that lead to fission product releases beyond the containment boundary.

This analysis is based upon the core-melt sequences that were identified and whose frequencies were estimated through the Accident Sequence Evaluation Program (ASEP) [9]. For each of the reference plants, a PRA has been performed previously. The Peach Bottom plant was the boiling water reactor (BWR) investigated in the Reactor Safety Study (RSS) [10]. The sequence frequency analysis from that study has been updated by ASEP to reflect the current plant configuration, accounting for changes that have been made to the plant since completion of the original study, and to incorporate improved risk assessment methods and data bases. A few planned, but not yet completely installed, plant modifications have been considered. These are modifications that have a specific time period by which they will be implemented by the utility and in some way are responsive to either recent rulemaking or safety concerns.

The ASEP analysis resulted in the identification of slightly less than two-thousand cut sets* which contributed to the overall core damage frequency. The base events which make up a cut set expression define exactly which systems in the plant have failed. Because each cut set represents a different combination of failures, it is theoretically possible for the containment response associated with each cut set to be different. Fortunately, this is not generally the case. It is possible to group the cut sets together such that each cut set within a group would have a similar containment response associated with it. For instance, two cut sets may differ in that for the first, diesel A was unavailable because it had been removed from service for maintenance, but in the second, diesel A was unavailable because it failed to start. The two cut sets would be grouped together, since for the containment analysis, the reason a diesel failed to run is generally irrelevant.

In this analysis (and in [9]), groups of cut sets which were formed to facilitate the containment response analysis are referred to as plant damage states. This term reflects the fact that each group has a specific level (or extent) of damage to the engineered safety systems of the plant. To give a precise definition, a plant damage state is a group of cut sets which presents a unique set of initial and boundary conditions to the containment analysis.

---

*A cut set is the minimal number of individual components, human and other, which will fail the systems required to prevent core damage.

Coordination between the cut set frequency analysis (ASEP) and accident progression analysis was accomplished through careful definition of plant damage states. Seven plant damage states were considered in the containment analysis. These are described in Section 2.3 of this report. Reference [9] includes a more detailed description of these damage states as well as a description of additional damage states which were not included in the containment analysis. Only plant damage states which were assessed (by ASEP) to have a mean frequency greater than $1x10^{-7}$/yr were considered in the containment analysis. It was judged that damage states with smaller frequencies would not significantly impact the overall results in terms of containment response given that core damage occurs.

The containment event tree developed by SARRP for Peach Bottom is considerably more complex than those considered in previous PRAs. It is in fact the most detailed of all the reference plant containment event trees. This is because BWRs typically have a significant level of system and phenomenological interactions between the containment and core cooling systems. The containment event tree for Peach Bottom is of sufficient detail to support an integrated assessment of uncertainty in the risk calculation, including modeling and data uncertainties in the assessment of sequence frequencies, containment loads, containment performance, operator intervention, and source term. The current tree includes consideration of repressurization of the reactor vessel as a result of containment pressurization as well as environmental effects on the operation of various safety systems. Previous PRAs have often neglected such effects or made simplifying assumptions. For instance, it has been common practice in prior PRAs to assume that all injection to the core would stop following containment failure. In the current treatment, the combination of events which could lead to this outcome (all injection precluded) is clearly identified. Thus, the analysts can isolate and understand underlying assumptions that produce a given result.

The following types of questions are in the Peach Bottom containment event tree:

(1) ENTRY STATES. What are the conditions in the reactor pressure vessel and containment prior to melting of the core that could influence the accident progression? These are provided primarily by the cut set group definitions that are discussed later.

(2) PHENOMENOLOGICAL EVENTS. What are the phenomena (e.g., direct containment heating or hydrogen combustion) that could affect the progression of severe accidents, at what points during the accident timeline do they occur, and what are their subsequent effects on the accident progression?

(3) REACTOR VESSEL BREACH CONSIDERATIONS. What is the size and location of the reactor vessel breach (if any) and the pressure in the system at the time of breach?

(4) CONTAINMENT SYSTEM SURVIVABILITY. Does the pressure suppression system survive the conditions occurring during severe accidents that exceed their design bases?

(5) CONTAINMENT FAILURE MODES (including the reactor building). What are the loads that challenge containment, does containment survive these loads, what is the nature of the failure (approximate size and location), and what is the subsequent pathway for fission product release to the environment?

The questions in the containment event tree are posed in ways that require the answers to be expressed in terms of likelihoods. For a station blackout accident in a BWR, for example, the likelihood that core degradation occurs under high- vs. low-pressure conditions in the primary system is considered. This has an effect on the energy released at vessel breach, which could affect the likely containment failure modes. Answers to such questions require information about the reactor design, the accident progression, the phenomenology of reactor accidents, and the capabilities of containment.

A single containment event tree has been developed to analyze the accident progression given many different initial conditions. The differences among the cut set groups are addressed by the first 16 questions in the tree, which set the initial conditions peculiar to each cut set group. Questions 17 through 107 are then used to investigate the specific possible accident progression pathways for each cut set group supplied by the ASEP analyses.

A large number of sources were utilized to obtain the needed information in order to construct and evaluate the containment event tree. These sources include References 1, 4, 5, 9, and 11 through 31.

The NRC provided general guidance with respect to the priorities to place upon the use of this variety of sources in evaluating the probabilities of the events considered in the containment event tree. Source term code package (STCP) calculations [4,5], to the extent possible, form the basis of the evaluations. The information developed by the Containment Loads and Performance Working Groups [11,12,13] was also considered. Where additional information was required, other analyses performed or sponsored by NRC were used. These sources were not sufficient to provide full treatment of some issues raised in the containment event tree. Thus, other information sources including the opinions of the SARRP expert review group [1] and analyses sponsored by the utility industry, were used. In addition, simple calculations were performed by the SARRP investigators. These are documented in Appendix B.

In quantifying the containment event tree, three separate evaluations, optimistic, central, and pessimistic, were propagated. These are intended to reflect the ranges both of possibility and of opinion among experts in the field of severe accident investigation. Thus, three sets of accident outcome probabilities were derived for each cut set group. The one labeled "pessimistic" tends to provide higher probabilities for the pathways that lead to higher source terms and lower probabilities for the lower source term pathways. The ones labeled "central" and "optimistic" are analogously interpreted. In this report weighting factors or distributions for these estimates are not proposed, and no attempt is made to determine whether one evaluation is more reasonable than another.* In this assessment, the central estimate analysis is presented in detail. The results of the optimistic and pessimistic estimates are only qualitatively discussed.

The central estimate is intended to represent the median of opinions in the reactor safety community on any particular issue. That is, if a large number of experts were to be polled with regard to their views, a substantial fraction would respond that the actual situation would be better than that indicated by the central estimate, and an approximately equal number would respond that it is worse. Of those who responded that the situation is actually better than that represented by the central estimate, it is expected that a substantial number would view the optimistic estimate as an appropriate representation of the issue. Likewise, those who viewed the situation as being worse than the central estimate implies, would agree that reality is, or could be, as unfavorable as that reflected by the pessimistic estimate. It should be pointed out that these principles were not rigorously applied in assessing some questions because, for some issues, there is a very small community of experts.

---

*The development of optimistic, central, and pessimistic pathways through the containment event tree provides a first step toward the evaluation of uncertainties in risk. In the SARRP risk report [1], the question of reasonableness is explored by developing and implementing a second procedure of uncertainty evaluation. The second procedure relies on a sampling of key issues over their ranges of belief using Latin Hypercube sampling methodology. The quantification of optimistic and pessimistic pathways in this report helps to establish the ranges of belief for the risk uncertainty evaluation. The sampling procedure (described in [1]) uses expert opinion to establish weighting factors for a selection of discrete levels within the ranges of belief.

The containment event analysis is based for the most part on the plant configuration that currently exists. There has been no attempt to account for plant changes that are planned but not yet implemented, with the exception that changes generally in response to recent rulemaking are included even if they have not yet been completely installed. For example, the symptomatic emergency procedures that include containment venting at 60 psig, alternate injection sources, and an automatic depressurization system (ADS) bypass switch soon to be installed to improve operator response in ATWS events have been included in this analysis.

2.2 Specific Example: Peach Bottom TBUX

The containment event tree for the Peach Bottom Mark I containment design includes a total of 107 questions. These questions cover all the accident sequences that were identified by ASEP as having frequencies of occurrence greater than 1×10^{-7} per year. A list of the questions together with reference to prior questions upon which the answers depend is provided in Table 2.1. The computer code input in Appendix E is specific to the "TBUX" cut set group (questions 1-16). This group is characterized by a long-term station blackout in which high-pressure core cooling is provided by steam turbine-driven systems for a period of about 6 hours, until battery depletion occurs. Failure of the steam-driven systems is then anticipated, and without power recovery, core melt begins about 4 hours later. This example is thoroughly treated in Appendices B through E.

2.3 Summary of Plant Features and Accident Sequences

In this section, specific plant features are discussed that significantly impact the nature of the expected response of the Peach Bottom Mark I containment to the accidents of interest. These features, in part, affect the form of the containment event tree. Also included in this section are descriptions of the various accident sequences that have been quantified for Peach Bottom.

2.3.1 Plant Features Important to Containment Response

There are several features of the reactor and containment designs in the Peach Bottom plant that tend to drive the containment response to severe accidents. Perhaps one of the most important factors is that the Mark I containment is a low-volume design since it uses a pressure suppression concept. The suppression pool contained in the torus is designed to condense the steam release in a large break LOCA. Under some severe accident conditions, however, the addition of noncondensibles due to the core degradation process or bypassing of the suppression pool may cause containment pressure rises beyond the design basis due to its low volume.

Another feature of Peach Bottom's containment is the fact that the containment is normally inerted (i.e., the oxygen concentration is maintained below 5 volume-percent by the addition of nitrogen). This eliminates any significant chance for hydrogen burns within the containment.

The BWR core support design (which provides individual support for each group of four fuel bundles from the vessel bottom head) is judged to minimize the probability of the pressurized ejection of core debris into the containment. Slumping of relatively small quantities of core debris (due to localized failure of the support) is anticipated to result in the depressurization of the vessel (due to local meltthrough) before large quantities of molten corium have collected in the bottom head. Direct heating is thus not a prominent cause of containment failure.

The drywell floor design is such that a water level no higher than 18 inches can be collected before water drains into the suppression pool. This relatively small amount of water significantly reduces the probability of large ex-vessel steam explosions. Associated with the drywell floor and pedestal design is the fact that radial flow of the core debris along the drywell floor would be unimpeded. This allows for attack of the drywell shell by the hot core debris, thereby creating a possible containment failure that bypasses the suppression pool.

Peach Bottom emergency procedures include containment venting procedures that make use of a series of possible vent paths for mitigating containment overpressure scenarios. The current procedures call for overriding containment isolation and venting at 60 psig. It is preferable to vent using a pathway through the suppression pool for fission product scrubbing. Since some vent paths could cause a release to the reactor building, particularly if the standby gas treatment system ductwork fails, the effects of venting on the response of the reactor building are examined.

One particularly important containment-mitigating system interaction is the modeled failure of the residual heat removal (RHR) system pumps when near-boiling conditions occur in the suppression pool. It is judged that these pumps at Peach Bottom, which provide suppression pool cooling or containment spray as well as other functions, will not function under saturated pool conditions (the pool serves as the suction source for the RHR pumps) due to insufficient net positive suction head (NPSH) for the pumps. While the high-pressure service water (HPSW) system pumps could be used through the RHR spray lines in an open-loop configuration, the ability to remove heat from the containment does not exist using this method for containment spray. Overpressure failure of the containment could therefore be delayed but not necessarily eliminated.

One other important feature is the relatively large pressure capability of the containment. The central estimate for this pressure failure point is 132 psia, based on an analysis performed by Ames Laboratory [26]. This pressure allows for an increased likelihood of successful venting in accident sequences characterized by a slow rise in containment pressure provided all other sufficient conditions exist for venting. On the other hand, this relatively high failure pressure hampers the ability to keep the primary system safety relief valves (SRVs) open against the containment backpressure. This interaction is important for an accident scenario in which the SRVs are required to be open to maintain low-pressure cooling of the core under conditions of very high containment pressures (e.g., such as when venting fails).

2.3.2 Description of Peach Bottom Plant Damage States

As stated previously, seven plant damage states were identified as being significant to the determination of the response of the Peach Bottom containment to postulated severe accidents. Each of these damage states is defined below. Multiple-letter designators are used to identify the plant damage states. The letters (or numbers) in the designator roughly indicate system failures associated with that damage state. The nomenclature used is similar to that used in WASH-1400, each character in the designator indicating a certain failure. The nomenclature is summarized in Table 2.2.

Table 2.3 indicates the mean frequencies of each of the plant damage states as assessed by ASEP. This information is useful in determining the relative importance of each damage state to the overall containment response to severe accidents.

TBUX:

This cut set group is characterized by transients leading to station blackout (loss of all ac power) as a result of coincident dc power failures. The loss of dc power causes failure of the diesels and the turbine-driven high-pressure cooling systems which results in the loss of all core and containment cooling. Without the restoration of offsite power in 30-40 minutes, primary system inventory boils off and core damage results. The core degradation process occurs under high-pressure conditions in the reactor vessel since the loss of dc power also disables operation of the ADS and SRVs.

A subset of this damage state (identified as TBUP in Reference 9) involves a stuck-open SRV. Based on ASEP results, the probability that an SRV will fail to reclose is approximately 5%, given a station blackout sequence involving loss of dc power. The damage state involving a stuck open SRV was considered in the containment analysis by assigning the branch point probability for a stuck-open SRV (Question 8 in the CET) to the value determined by ASEP. This subset of TBUX is similar to the TBUP plant damage state (described below), in that Core damage occurs with the vessel depressurized,* but differs in that the failure of dc power reduces the likelihood that offsite power can be recovered (discussed in Appendix B). It is thus treated as part of the TBUX damage state.

TBU:

This cut set group is very similar to TBUX except that there are no dc common mode failures. The steam-turbine-driven HPCI and RCIC pumps are unavailable as a result of random failures. This latter distinction allows for vessel depressurization and, thus, core damage is likely to occur under depressurized conditions in the reactor vessel. This cut set group along with TBUX collectively makes up all the dominant short-term core damage scenarios due to station blackout.

TBUP is a subset of this plant damage state which includes a stuck-open SRV. However, because of its low frequency (~5% of the TBU frequency) and the fact that the reactor vessel is most likely depressurized during core damage for this damage state, even without a stuck-open SRV, the distinction of this subset is essentially moot.

TB:

This cut set group is characterized by transients leading to a long-term station blackout (loss of all ac power). Core cooling is successful with turbine-driven high-pressure cooling systems providing coolant injection until about 6 hours into the scenario. At that time, the batteries deplete, affecting the ability to continue operation of these systems. Without ac power recovery within 4 hours of battery depletion, core damage results. While the primary system may be initially at relatively low pressures, depletion of the batteries causes loss of ADS and SRV control. The vessel will therefore repressurize with core damage progressing under high-pressure conditions.

---

*A stuck-open SRV allows depressurization of the reactor vessel even if the SRVs cannot otherwise be opened in relief mode.

As with the TBUX damage state, some subset of the TB damage state involves the failure of an SRV to reclose. This cut set group (referred to as TBP in Reference 9) was treated as part of the TB cut set group in the containment analysis. This was accomplished (as for the TBUX cut set group) by using the split fraction between these two groups, determined by ASEP, as the branch point probability for a stuck-open SRV in the CET (Question No. 8). Again, the TBP subset results in the reactor vessel being depressurized during core damage which is otherwise highly unlikely for TB.

TCSX:

The TCSX cut set group is characterized by an ATWS with either main steam isolation valve (MSIV) closure or with the MSIVs initially open but subsequently closing. This isolates the primary system under high-power conditions, thereby rapidly increasing the pressure and temperature within containment since RHR cooling under these circumstances is inadequate. Timely start of the standby liquid control system (SLCS) is not performed or it is assumed to fail as a result of being left in an inappropriate configuration after the last test of the system. Core cooling is maintained for a short time (~1/4 hour) before HPCI fails due to high pool temperature. The operator then fails to rapidly depressurize the RPV (so that low-pressure systems can be used), which leads to core damage. ASEP assumed that the high-pressure coolant injection (HPCI) pump would fail when the temperature of the water in the suppression pool exceeded the pump qualification temperature. Thus, this damage state progresses to core damage with the RPV at high pressure and the containment pressurized (due to the elevated pool temperature) to between 30 and 50 psig.

TCUX:

The TCUX cut set group is also characterized by an ATWS with either MSIV closure or with the MSIVs initially open but subsequently closing. This isolates the primary system thereby rapidly increasing the pressure and temperature conditions within containment since RHR cooling under these circumstances is inadequate. The SLC system is started by ~4 minutes into the accident, but initial high-pressure coolant injection (HPCI) failure under high-power conditions and operator failure to rapidly depressurize the vessel (so that low-pressure systems can be used immediately) lead to a quick core damage event.

TCSRX2:

This cut set group is characterized by an ATWS with either MSIV closure or with MSIVs initially open but subsequently closing. This isolates the primary system under high-power conditions thereby rapidly increasing the pressure and temperature conditions within containment since RHR cooling under these circumstances is inadequate. Timely SLC system start is not performed or it fails due to being left in an inappropriate configuration after the last test of the system. Core cooling is maintained first by HPCI and then by low-pressure cooling systems after HPCI failure due to a high pool temperature in ~15 minutes. Venting of the containment is not successful initially. Containment failure results. This damage state results when the induced failure of the containment is not sufficiently large (in terms of effective flow area) to relieve the steam pressure within (i.e., leak failure rather than rupture). Such a failure precludes continued operation of the low-pressure cooling systems. This is because the maximum nitrogen pressure to the SRVs is ~100-125 psi, which is under the estimated 130+ psi pressure for containment failure. The differential pressure between the nitrogen system and the containment is insufficient to keep the SRVs open. Therefore the vessel repressurizes and all core cooling is lost.

TCSRV23:

This cut set group is very similar to TCSRX2. It is differentiated in that containment failure is a rupture rather than a leak. Since the suppression pool is saturated following containment failure, the low-pressure ECCS is inoperable due to lack of net positive suction head. In this damage state, the operators fail to restore injection, using condensate or high-pressure service water, prior to core damage. The central estimate frequency of this damage state is less than 1×10^{-7} per year. However, in the SARRP study of uncertainties [1] contribution from this damage state becomes significant under some sets of assumptions.

3. RESULTS

The results of the containment response analysis are presented in this section. The containment response is described in terms of how the accident progresses from a given plant damage state. Strictly defined, the accident progression is the set of 107 answers to all the questions in the CET. In other words, an accident progression is a pathway through the branch points in the event tree. There are literally tens of thousands of accident progressions for each plant damage state analyzed. It was therefore necessary to bin (or group) these progression according to features which were relevant to the mode of containment failure. An accident progression bin is thus a group of accident progressions for which the containment response is similar. (The grouping of accident progressions into bins is analogous to the grouping of cut sets into plant damage states.)

Table 3.1 summarizes the results of the central estimate analysis for each plant damage state. The results are given in terms of progression bins which indicate the features of the containment response expected to have the largest impact on fission product release and ex-plant consequences. The features of the accident progression which define these bins are as follows:

- Suppression pool subcooled or boiling;
- Timing, location, and size of containment failure;
- Potential for fission product retention in the reactor building;
- Whether or not the reactor vessel is breached;
- Whether or not core-concrete interaction (CCI) occurs;
- Operation of drywell sprays.

The assessed conditional probability of each bin is indicated in the table. Only bins with a conditional probability greater than 0.01 are shown. If a bin has a probability less than 0.01 for a particular damage state, a dashed line is entered in the table. Tables 3.2 and 3.3 summarize the results for the optimistic and pessimistic estimates in the same manner.

Figures 3.1 through 3.4 present the accident progression results in a graphical display. Only bins having the largest conditional probability are displayed. The bin numbers indicated in Table 3.1 provide a key to the bin numbers indicated in the figures.

Appendix A provides the full detail of the central estimate results. The binning of the accident progressions in Appendix A preserves sufficient detail in the results to support any anticipated future use, including the interface to the risk analysis [1]. While it is probably unnecessary to provide such detail, it was judged preferable to preserve this information rather than have to repeat the analysis if some future work required information not preserved in the binning presented in Table 3.1. For instance, from Table 3.1 the approximate probability of drywell failure at vessel breach can be determined. However, the mode of failure (meltthrough of the shell or overpressurization) cannot be determined. This detailed information can be determined from the tables in Appendix A.

The remaining discussion in this section provides important insights obtained by the analysts based on a thorough study of the results in Appendix A. The numerical results were derived from that information and thus are more accurate than the summary provided in Table 3.1. As a simple example of the approach used, consider the question of drywell meltthrough for the TBUX plant damage state. From Table A-1, drywell meltthrough is indicated by the designator "CFD-IM." This term in column 3 of the subsequent tables in Appendix A indicates that containment failure was induced by meltthrough of the drywell wall. For example, Table A-4 presents the results for TBUX. Summing all the probabilities in column 1 of Table A-4 for progression bins with the "CFD-IM" designator in column 3 gives a value of 0.45. Thus, the conditional probability of containment failure by drywell wall meltthrough for the TBUX damage state is 0.45.

Plant damage states involving station blackout constitute approximately 90% of the core damage frequency for Peach Bottom. Because of this predominance of blackout, recovery of offsite power has a potentially large impact on the overall containment performance for severe accidents. Thus, the containment analysis includes provision for recovery of power both before and after reactor vessel breach. This consideration is included in the results presented in this section.

In a like manner, recovery is considered for the ATWS-initiated damage states involving initial failure of the operators to depressurize the RPV. For these damage states (TCUX and TCSX) the low-pressure emergency core cooling (ECC) pumps are available. Depressurization of the RPV would result in automatic coolant injection. The possibility that the operators belatedly remove the automatic depressurization system (ADS) from its inhibited condition is recognized in the analysis. The results below reflect this consideration.

3.1 Cut Set Groups TBUX and TBU

These two cut set groups collectively represent all of the dominant short-term station blackout scenarios with early loss of all core cooling as well as containment cooling. The primary difference between the two cut set groups is that the reactor vessel is at high pressure during the beginning of core degradation in TBUX. If an SRV has failed to reclose (TBUXP subset of the TBUX damage state) then slow depressurization of the RPV is assumed to occur during core degradation. TBU allows for controlled depressurization depending on operator response.

The central estimate probabilities (conditional on core melt) for the containment failure modes and resulting accident progression pathways are nearly identical for all three cut set groups as shown in Table 3.1 and Figure 3.1.

The dominant mode of containment failure for these two cut set groups is drywell rupture (leakage area large enough to depressurize the containment) at vessel breach. For the TBUX cut set group, there is a 64% probability of this mode of failure. Meltthrough of the drywell shell constitutes 70% of that fraction (0.45 probability conditional on core melt). For the TBU cut set group, drywell rupture at vessel breach has a conditional probability of 0.46. Essentially 100% of this probability is a result of the drywell wall meltthrough mode of failure. Failure due to pressurization of the containment as a result of vessel breach is very unlikely for this cut set group because the operators are able to depressurize the RPV.

The TBU results are, of course, very sensitive to the assumed probability of drywell meltthrough. However, if the RPV is not depressurized, containment failure as a result of overpressurization at vessel breach is likely. Thus, for TBUX, if the drywell wall meltthrough probability were reduced, it is not anticipated that the probability of "early" failure (i.e., before core-concrete interaction begins) would decrease significantly. It is noted that this conclusion depends on the assumed value of containment failure pressure (132 psia). However, the containment capability that would be required to avert failure as a result of RPV blowdown was not assessed in this study. Also, the sensitivity of these results to quantity of hydrogen generated during core damage was not investigated.

For both of these cut set groups the second most likely result, in terms of containment performance, is no loss of integrity. For TBUX there is approximately 0.1 probability that the containment remains intact. This probability is approximately 0.2 for the TBU cut set group. The difference between these two is a direct consequence of the reduction in the probability of recovering ac-power without dc-power being available (as it is not for TBUX).

Intentional venting of the containment from the wetwell airspace contributes significantly to the loss of integrity. Because venting was assessed to be very difficult without ac-power, the large majority of accident progressions which include venting involve the venting occurring after vessel breach and following the restoration of offsite power. Venting prior to vessel breach is unlikely for these cut set groups since the containment is unlikely to attain the 60 psig at which the plant procedures instruct venting to be initiated. For the TBUX cut set group the probability that the containment would be vented late was assessed to be 0.075. This value was 0.12 for the TBU cut set group.

If the release pathway for fission products is through the suppression pool, environmental release will be reduced by the retention of fission products in the suppression pool. For the TBUX cut set group the probability that the majority of the release will pass through the suppression pool is 0.081. Four percent of that fraction (0.004 conditional probability) involves venting of the wetwell after a drywell leak has developed. The drywell leak is caused by the blowdown from the RPV at vessel breach but is not large enough to depressurize the containment. For the TBU cut set group the probability is 0.13 that leakage occurs only from the wetwell. All but a small fraction of that probability is attributed to intentional venting.

Recovery of offsite power impacts the accident progression in several ways. Allowing for containment venting is one impact. However, the restoration of coolant flow can arrest further melting of the fuel. The probability of preventing breach of the RPV was assessed to be 0.084 for TBUX and 0.14 for TBU. The probability of arresting the accident subsequent to vessel breach was 0.072 and 0.078 for TBUX and TBU respectively. For those accident progressions in which core-concrete interactions did occur, drywell sprays were assessed to be available for 14% in TBUX and 28% in TBU.

Source term calculations indicate that the reactor building can play an important role in fission product retention. However, the combustion of hydrogen and/or carbon monoxide within this structure has the potential to compromise this retention by inducing structural damage. In the central estimate CET, approximately 20% probability is assigned to such induced bypass of the reactor building, given that combustion occurs. (Given failure of containment, combustion is very likely.) For TBUX accident progressions the reactor building is bypassed during the periods of significant release (leakage from containment without suppression pool scrubbing) with a 0.21 probability. This value is 0.18 for the TBU cut set group.

The results for the optimistic walkthrough are summarized in Table 3.2. The bin numbers in Table 3.2 (and 3.3) correspond to the numbers in Table 3.1 (i.e., the same bin has the same number). Bins not occurring in the central estimate are numbered consecutively starting at 105. (The last bin in Table 3.1 is 104.) There are fewer bins in the optimistic (and pessimistic) estimates than in the central, because a significantly larger probability was assigned to particular outcomes for several questions in the CET. The central estimate often assigns a significant probability to more than one outcome, thus creating more pathways through the tree which have significant probability.

The predominance of the in-vessel recovery scenario is evident. This probability is 0.33 for TBUX and 0.42 for TBU. As in the central estimate, this difference is governed by differing probabilities of offsite power recovery. In fact, the difference between the assumed power recovery probability is more clearly pronounced in the optimistic walkthrough results. The TBU results show a 0.42 probability that the core debris will be quenched on the drywell floor following vessel breach with the containment remaining intact. For TBUX, however, it is assessed that the containment pressurization which accompanies vessel breach makes wetwell venting the likely result of these late recovery scenarios. The optimistic results show a substantial contribution from scenarios involving late venting of the wetwell without recovery of ac-power. Because venting without power was judged to be both difficult and hazardous to the operations staff, these accident progressions do not have significant probability in the central estimate. For TBUX, the optimistic estimate has a probability of 0.22 for late successful venting. This probability is 0.16 for TBU, reflecting again that recovery of offsite power (arresting the accident) is more likely for this cut set group.

For the pessimistic estimate, the accident progressions which dominate the central estimate are far more likely. For TBUX the probability of drywell rupture at vessel breach is 0.96. This probability is 0.72 for TBU. In both cases the most likely mode of failure is meltthrough of the drywell wall. However, for TBUX, in which the RPV is pressurized at the time of breach, drywell rupture as a result of over-pressurization is almost equally likely. Scenarios with some form of recovery make up the remainder of the TBU accident progressions. All involve loss of containment integrity. There is a 0.15 probability that the containment will be vented early through the wetwell.

3.2 Cut Set Group TB

One cut set group represents all of the dominant long-term station blackout scenarios with initial core cooling until battery power depletes about six hours into the event. Cut sets involving a stuck-open SRV (TBP) that keeps the reactor vessel depressurized are included in these results. In TB, if there is no stuck-open SRV, the vessel repressurizes leading to core degradation under high-pressure conditions unless power recovery is successful.

As for the short-term blackout plant damage states (TBUX and TBU), rupture of the drywell at vessel breach is the dominant mode of containment failure. The conditional probability is

0.44. Meltthrough of the drywell shell constitutes 24% of that probability. Overpressure failures of the containment are more likely for this damage state (total probability 0.39) than for the short-term blackout owing to the elevated containment pressure (at least 13 psig which results from heatup of the suppression pool during HPCI or RCIC operation) at the time core melt begins. As for the TBUX damage state, in the absence of ac-power recovery, the RPV is pressurized at the time of vessel breach because dc-power is unavailable.

As a consequence of the relatively long time delay (~10 hours) before the onset of core damage, venting of the containment through the wetwell prior to core damage is a significant possibility. This is the second most likely outcome in the analysis. The assessed probability is 0.27. For 27% of those cases the drywell shell is melted subsequent to vessel breach and thus the venting is not successful in terms of mitigating fission product releases.

The third most likely outcome (0.15 probability) is that the accident is arrested with the containment intact. Eighty-four percent of these accident progressions involve recovery prior to vessel breach. Thus, given that the RPV is failed, it is unlikely for the TB cut set group that containment integrity can be maintained.

Late venting of the containment is successfully accomplished with an assessed probability of 0.061. This is the most likely outcome if the containment has survived breach of the RPV and offsite power has been recovered. Other modes of late failure are very unlikely (less than 0.02 total probability).

Release pathways involving passage through the wetwell are more likely than for the short-term blackout. The assessed conditional probability is 0.27. This difference between the two damage states is attributable mainly to the increased probability for wetwell venting, which contributes 93% to the probability that the torus is the location of leakage.

Successful operation of the drywell sprays was assessed to occur for 22% of the accident progressions involving core-concrete interaction. This fraction is larger than for the TBUX damage state, reflecting greater probabilities of offsite power recovery but lower than the TBU damage since the suppression pool is saturated for TB and the drywell spray mode of RHR is unavailable. Accident progressions with reactor building bypass during periods when significant release from the primary containment are occurring have a probability of 0.15 in the central estimate. This probability is somewhat lower than for the short-term blackout scenarios, owing to the increased probability (for TB) that the containment will be successfully vented.

Table 3.2 summarizes the results from the optimistic estimate. The most likely outcome for the optimistic estimate is that both the containment and the RPV remain intact (0.74 probability). This estimate relies on a relatively high probability for the restoration of offsite power given the long time available. The second most likely outcome is that the containment will be vented. This outcome has an assessed probability of 0.1. Even if the vessel is breached, the optimistic assumptions allow a high probability for quenching of core debris on the drywell floor. Thus, the total probability of maintaining containment integrity for the optimistic case is 0.82.

The pessimistic results are summarized in Table 3.3. Relatively early failures of the drywell predominate these results. The assessed probability of drywell failure as a result of vessel breach is 0.57. Venting of the containment during core damage is assessed to be relatively likely (0.35 probability). However, in essentially all cases in which early venting occurs, the drywell shell is subsequently melted and suppression pool bypass ensues (0.34 probability).

3.3 Cut Set Group TCUX

This cut set group represents ATWS events followed by the loss of all core cooling. The RPV is initially at high pressure because the operators inhibit ADS. The results of the CET analysis are very sensitive to the assumed probability that the operators belatedly depressurize the vessel (i.e. after core damage).

As for most of the damage states analyzed, overpressure failure of the containment, which results from the blowdown of the RPV at vessel breach, is the most likely outcome. The central estimate probability of this failure mode is assessed to be 0.5 (i.e. essentially certain provided the RPV is not depressurized). The predicted large increase in containment pressure during core damage, which leads to this failure mode, is a consequence of the large quantity of hydrogen which is predicted to be generated by the Source Term Code Package [4,5] when water is supplied to the RPV by the control rod drive (CRD) hydraulic system. For the ATWS cut set groups, continued operation of the CRD pumps during core damage was assessed to be highly likely. For 13% of these failures, the containment only leaks and subsequent venting of the wetwell assures that the dominant pathway for fission product release is through the suppression pool. Of the remaining fraction, 93% involve ruptures of the drywell.

The second most likely outcome is that the containment remains intact. The probability of this result is 0.27. For 85% of these accident progressions, the RPV also remains intact.

Late venting of the containment is the third most likely outcome, having an assessed probability of 0.16. Early venting (i.e. prior to vessel breach) is highly unlikely because the containment does not attain the 60 psig pressure which would lead to the initiation of venting procedures until after RPV meltthrough.

Meltthrough of the drywell shell subsequent to vessel breach is assessed to occur with a probability of 0.13. As for the damage states discussed earlier in this section, this mode of containment failure would be more significant if the assessed failure pressure of the containment were larger.

Release pathways involving the wetwell are assessed to have a conditional probability of 0.20. Intentional venting accounts for 82% of this probability.

Since ac-power is always available in this cut set group, and because the suppression pool is subcooled, operation of drywell sprays is highly likely. (Spray water is taken from the suppression pool and if the pool is subcooled any problems associated with insufficient suction head are avoided.) Drywell sprays operate for essentially 100% of the accident progressions involving core-concrete interaction.

Reactor building bypass is assessed to occur for 13% of the accident progressions which would involve significant releases from the primary containment. The assessed probability of such scenarios is 0.067. Venting of the containment and the substantial probability of arresting the accident result in this low contribution.

The optimistic results for this accident are easily summarized. The probability is 0.89 that the containment integrity is retained. The remaining accident progressions involve containment venting but the operation of drywell sprays is assumed to preclude core-concrete interaction.

The pessimistic results are similarly aligned toward a single outcome. This outcome is containment failure at vessel breach (0.96 probability) reflecting the pessimistic assumption that the operators, having inhibited ADS, will (very probably) not reconsider lowering RPV pressure. Most of the remaining accident progressions (0.03 probability) involve early venting of containment followed by meltthrough of the drywell shell by core debris subsequent to RPV breach.

3.4 Cut Set Group TCSX

This cut set group is very similar to TCUX with the exception that HPCI provides core cooling until the suppression pool temperature reaches approximately 250 F. At this point core cooling is lost. The operators fail to depressurize the RPV prior to core damage. The accident proceeds as for TCUX but the containment pressure is larger at the onset of core damage as a result of the suppression pool heatup during HPCI operation.

Unlike TCUX, venting of the containment during core damage is the most likely mode for loss of containment integrity. Early venting occurs with an assessed (central estimate) probability of 0.57. Venting is more likely here because the pressure threshold for initiation of the venting procedure is exceeded during core damage. However, in 48% of these cases meltthrough of the drywell shell subsequent to vessel breach results in complete bypass of the suppression pool.

As for TCUX, the second most likely outcome is that the containment remains intact. The probability is lower (0.16) owing to the increased probability of either venting or containment failure which results from a higher initial containment pressure.

Failure of the containment at, or shortly after, vessel breach is actually more probable than the retention of integrity (0.19 probability). However, two mechanisms contribute to this. Overpressure failure has a central estimate probability of 0.1. The probability that the drywell shell will melt through is assessed at 0.081 (0.38 if one includes accident progression in which venting occurs prior to vessel breach).

Late failures of the containment have an aggregate probability of 0.052. Venting subsequent to vessel breach constitutes almost 93% of this result.

Because containment venting has a substantial probability of success, release pathways involving the wetwell have an assessed probability of 0.33. Mitigation of fission product releases is also accomplished through the operation of drywell sprays. The suppression pool is saturated and thus the service water system must be used to provide spray. This is less likely to be used than the residual heat removal (RHR) sprays and thus only 55% of the accident progressions involving core-concrete interactions also involve drywell sprays.

Reactor building bypass is assessed to occur for 25% of the accident progressions which involve significant releases (i.e. releases from the drywell). The assessed probability of such scenarios is 0.12.

Under optimistic assumptions, the rise in containment pressure resulting from in-vessel hydrogen generation is small relative to the containment failure pressure (a few percent). Thus TCSX is very similar to TCUX in terms of the optimistic results for containment performance. Table 3.2 shows the results to be identical.

The pessimistic results are also similar to TCUX. The probability of drywell failure at vessel breach is essentially unity owing to the larger pressure in containment at the onset of core damage. Because the difference between the TCUX and TCSX scenarios is a matter of approximately 30 psig difference in containment pressure, it should be expected that the results of the optimistic and pessimistic sets of assumptions are similar.

3.5 Cut Set Group TCSRX2

This cut set group, by definition, includes a leakage failure of the containment prior to core damage. Thus, core damage begins with the RPV at high pressure since the containment pressure is too large to allow operation of the SRVs in relief mode.

At core uncovery, the location of containment leakage is divided as follows: 83% drywell head seal, 12% other locations in the drywell, 5% torus. It is assumed that the drywell head leakage results from essentially elastic deformation of the closure bolts. Thus, if the containment is vented during core damage the containment will depressurize and the vent pathway becomes the dominant release pathway (at least temporarily). If the initial leak location is not the drywell head, the size of the hole in containment must be sufficient to pass all the steam generated during the ATWS. It is assumed that this hole size is sufficiently large to allow the containment to rapidly depressurize (within 30 minutes) once the reactor core uncovers. (Core uncovery occurs rapidly once injection of the RPV ceases.)

The most likely final pathway for leakage from the containment is from the drywell. Large leakage (i.e. a rupture) is assessed to have a probability of 0.69. Leakage at a slow rate (i.e. the containment remaining pressurized) is assessed to have a probability of 0.13. This latter class of outcomes involves only leakage from the head seal. Meltthrough of the drywell shell accounts for 89% of the drywell rupture accident progressions. This includes accident progressions having an aggregate probability of 0.49 in which the containment is intentionally vented. In 65% of those accident progressions in which venting occurs, subsequent meltthrough of the drywell results in suppression pool bypass.

Reactor building bypass is assessed to occur for 24% of the accident progressions involving leakage from the drywell. The total probability of accident progression involving drywell leakage and reactor building bypass is 0.20 (which is essentially the central estimate assumption). Bypass is more likely since drywell head leakage occurs directly to the refueling bay and hydrogen combustion in the refueling bay is judged to have a high probability of resulting in complete bypass.

Core-concrete interactions occur with a probability of 0.52. In the remaining accident progressions the core debris is quenched ex-vessel. Drywell sprays operate for 49% of those accident progressions involving core-concrete interaction. Because the suppression pool is saturated (containment integrity not maintained) the emergency core cooling pumps are assumed unavailable during core damage and thus recovery in-vessel has negligible probability. This assessment is based on consideration of the time required to vent containment (thus allowing the SRVs to reopen) plus the time required to initiate injection with either condensate or HPSW.

The optimistic results involve a substantial probability (0.95) that the primary release pathway for fission products will be through the suppression pool. This result is driven by the optimistic assumption that venting would be successful even if the initial location of leakage was the drywell.

The pessimistic results show essential certainty that releases being from the drywell will be the dominant mode of containment failure. This result stems partially from the pessimistic assumption that the drywell is structurally weaker than the torus (and is thus the dominant location for failure). In addition, even if venting is initially successful, meltthrough of the drywell shell by core debris results in drywell leakage being the path of least resistance.

3.6 Cut Set Group TCSRV23

The central estimate frequency of cut sets in this group is sufficiently low (2.5 x 10^{-8}/yr) that it would not be considered since a 1 x 10^{-7}/yr cut-off has been applied. However, in the Limited Latin Hypercube uncertainty analysis [1], the frequency of this cut set group was higher than the cut-off for certain sets of assumptions. Thus, the central estimate results for this cut set group are included herein for reference.

TCSRV23 is initiated in a manner identical to TCSRX2. The difference lies in the mode of containment failure. For TCSRV23 the containment depressurizes as a result of failure. With the containment depressurized, the RPV will depressurize (since it was initially depressurized). Core damage results only if the operators fail to restore injection following containment failure. A low probability is assigned to such failure on the part of the operators. However, if core damage does occur, it occurs in an open containment.

All accident progressions involve rupture of containment prior to core damage (this is part of the cut set group definition). The probability of drywell rupture is 0.73. The probability of torus rupture is 0.27. For 38% of the accident progressions involving initial wetwell failure, subsequent meltthrough of the drywell shell by core debris results in the bypass of the suppression pool.

Core-concrete interaction is prevented with a probability of 0.56. For the remaining accident progressions, in which core-concrete interactions occur, drywell spray is available in 48%. As for TCSRX2, it was judged that the prevention of RPV breach had negligible probability.

Reactor building bypass, for accident progressions in which the dominant release pathway is directly from the drywell, occurs for 30% of the accident progressions involving leakage from the drywell. The total probability for such accident progressions is 0.25.

The optimistic results indicate virtual certainty that containment failure occur in the torus. In addition, because condensate and service water pumps are available, the core debris is assessed to be quenched ex-vessel. (The optimistic assumptions include coolability of core debris.) Thus, this accident is relatively benign for the optimistic estimate.

The pessimistic results indicate essential certainty of drywell failure. In fact, one accident progression bin accounts for almost 93% of the cut set group frequency. The remaining accident progression bins involves an ex-vessel steam explosion, following vessel breach, which is assumed to disperse the core debris sufficiently that drywell sprays will cool it.

4. SUMMARY AND CONCLUSIONS

To interpret the results in the most illuminating way, it is necessary to restate the meanings of the words "optimistic," "central," and "pessimistic." The central estimates are intended to provide a glimpse of the median of the reactor safety community. That is to say, there is a reasonable expectation that if the community were polled on the subjects treated in this document, a substantial fraction would respond that the reality of the situation is better than that depicted by the central estimates, and a comparably substantial fraction would respond that it is worse. Among those who think the reality is better (a group referred to as "optimists"), a substantial fraction would claim that the optimistic estimates apply. Similarly, many of the opposing group (the "pessimists") would claim that reality is or could be as unfavorable as the pessimistic estimates imply. Again, it should be pointed out that it was not possible to apply these guidelines rigorously as a consequence of the small number of experts whose views were available on particular issues.

Viewed in this context, the results of the containment event analysis indicate quite clearly that there are significant uncertainties in the reactor safety community's understanding of containment loading and response. The differences between the optimistic and pessimistic results attest that for many accident sequences, one cannot state the probabilities of the containment failure modes within narrow limits. However, one can identify the factors that drive the results and understand the reasons for the differences.

The most important conclusion of this analysis is that, given severe fuel damage occurs, there is a substantial probability of drywell failure. Containment failure in the drywell is particularly significant because fission product aerosols escaping the containment from the drywell bypass the suppression pool. If containment failure were to occur in the torus, in the airspace above the suppression pool, aerosols escaping the containment would most likely pass through the suppression pool wherein a large fraction would be retained, as a result of scrubbing.

Timing is an important aspect of these results with regard to drywell failure. The majority of containment failures occur at, or shortly after, RPV failure. This timing may have significance with regard to the offsite consequences of the accident. In the consequence calculations reported in Reference 1 it is assumed that the offsite population begins to evacuate approximately one hour after core uncovery. RPV failure is estimated to occur one-and-a-half to three hours after core uncovery. Thus, even if containment failure occurs at the time of RPV failure, which is likely according to the current results, the offsite population will evacuate in advance of radioactive material reaching their initial locations. However, these results offer little margin of safety with respect to evacuation timing, particularly for inhabitants close to the site boundary.

It must be recognized that the above conclusions are highly sensitive to assumptions in the central estimate analysis which may reflect a degree of conservatism. The key determinants here are: 1) structural failure of the containment due to overpressurization will occur in the drywell; 2) meltthrough of the drywell shell has a significant probability and would occur shortly after vessel breach; 3) the mean failure pressure of the containment is 132 psia. The assumptions with regard to containment failure pressure and location were judged by the SARRP reviewers [6,7,8] to probably be conservative (i.e. to lead to results which had a pessimistic leaning). However, these assumptions are based on the most recent analysis currently available. It would not be reasonable for the SARRP analysts to propose some other failure pressure and location based only on the knowledge that the results of the available analysis are based on conservative modeling and failure criteria. It was beyond the scope of the current work to perform new analyses of this type.

The assumptions with regard to drywell wall meltthrough have similar origins. The only available analyses suggest drywell wall meltthrough will occur immediately after RPV meltthrough. There are assumptions made in those analyses which are clearly open to question. The SARRP analysts have proposed alternate scenarios (described in Appendix B) which would suggest that meltthrough could be delayed several hours or may not occur at all.

Because station blackout is such an important sequence for Peach Bottom (in terms of frequency) it is not surprising that accident progressions involving offsite power recovery are significant in the results. Power recovery leads to features in the accident progression which would serve to mitigate fission product releases to the environment: 1) quenching of core debris (preventing RPV meltthrough in some instances); 2) containment venting; 3) operation of drywell sprays.

The optimistic and pessimistic results reflect, in many respects, the results obtained when bounds of the underlying assumptions are utilized. In particular, the pessimistic results show higher probabilities of early drywell failure due to the assumptions made with regard the quantity of hydrogen generated in-vessel and the attendant containment pressure increase. An assumed high probability for meltthrough of the drywell wall is also an important contributor. The optimistic results show negligible probability of early drywell failure and relatively large probability of recovery prior to containment failure. Low in-vessel hydrogen production, combined with a higher value for the containment failure pressure, leads to the result that if the containment fails at all, it fails late. The dominance of scenarios without containment failure in the optimistic estimate results from the assumed high probability for coolability of the core debris. The nonrecovery probability for offsite power is also lower than in the central estimate.

Table 2.1

SUMMARY OF CET EVENTS FOR PEACH BOTTOM

Questions Posed by the Event Tree	Dependencies on Prior Questions
1. What is the initiating event?	None
2. What is the initial break location?	None
3. What is the level of pre-existing leakage or isolation failure	None
4. Is there a loss of offsite power?	None
5. Is there a station blackout (diesel generators fail)?	None
6. Is dc power not available?	None
7. For TC, does SLC fail to inject?	None
8. Does a S/RV stick open early?	None
9. Do the HPCI and RCIC systems fail to inject?	None
10. Does the CRD hydraulic system inject?	None
11. Do the LPCS and LPCI systems fail?	None
12. Do the RHR systems fail?	None
13. Does the condensate system fail?	None
14. Does HPSW fail in a mode which would preclude injection?	None
15. Is there a failure which precludes drywell cooler operation?	None
16. Is ADS blocked or not called upon?	None
17. Does the RPV remain pressurized?	1, 5, 6, 8, 9, 10, 11, 12, 16
18. For TC, is inadequate level maintained with LPI?	1, 7, 16, 17
19. What type of sequence is this (summary of plant damage)?	1, 5, 9, 10, 11, 12, 14, 17, 18
20. Is there an LP system break induced by power cycling (ATWS)?	1, 16, 19
21. Do low-pressure ECC systems continue/begin to inject?	11, 17, 19
22. Do containment sprays fail before core degradation?	1, 11, 12, 14, 19, 21
23. Is the containment vented before core degradation?	1, 19, 20
24. Does the RPV repressurize before core damage?	8, 17, 19, 20, 23
25. Does (do) any S/RV-tailpipe vacuum breaker(s) stick open?	8, 17, 19, 24
26. Is there a repressurization break of low-pressure systems?	24
27. Does containment fail before core degradation?	19, 20, 23, 26
28. What is location of early containment leakage?	19, 20, 23, 26, 27
29. For drywell failures, is the failure at the head seal?	19, 20, 26, 28
30. What is the containment leakage level before core degradation?	1, 19, 20, 23, 26, 27, 28, 29
31. Is the suppression pool drained before core degradation?	27, 28, 30

Table 2.1 (Continued)

SUMMARY OF CET EVENTS FOR PEACH BOTTOM

Question Posed by the Event Tree	Dependencies on Prior Questions
32. Does ac power remain lost during core degradation?	5, 6, 19, 24
33. What is the status of CRD?	5, 10, 23, 30, 32
34. What is the status of HPSW?	5, 14, 23, 30, 32
35. Does the core melt?	13, 17, 19, 24, 26 30, 33, 34
36. What is containment pressure before core degradation?	1, 9, 12, 19, 20 22, 26, 30
37. Will the suppression pool flash following containment vent or rupture?	1, 8, 9, 12, 19, 21, 24
38. Is the RPV depressurized during core degradation?	5, 17, 19, 20, 21, 23, 24, 26, 30, 32, 37
39. Is there injection during core degradation?	13, 21, 30, 31, 32 33, 34, 37, 38
40. What is the status of containment sprays?	21, 22, 30, 31, 32, 37
41. What is the level of flow to the drywell during core degradation?	1, 20, 25, 26, 38, 39, 41
42. What is the hydrogen generation and pressure rise during core degradation?	1, 17, 19, 20, 23, 24, 26, 30, 33, 38, 39, 41
43. Does at least one drywell vacuum breaker stick open?	1, 17, 19, 20, 24 25, 26, 28, 42
44. Does hydrogen discharge lead to suppression pool bypass?	17, 24, 42
45. Does hydrogen discharge to suppression pool induce wetwell failure?	17, 24, 44
46. Is the vent threshold reached during core degradation?	45
47. Does containment venting not occur during core degradation?	5, 12, 19, 23, 30, 32, 40, 42, 46
48. Does the RPV repressurize during core degradation?	1, 8, 38, 45, 47
49. Is there a repressurization break during core degradation?	48
50. Does the containment fail by pressure during core degradation?	19, 29, 30, 45, 47, 49
51. What is location of containment leakage prior to vessel breach?	28, 30, 45, 47, 49, 50
52. For drywell failures, is the failure at the head seal?	29, 30, 47, 51
53. What is the level of containment leakage before vessel breach?	19, 29, 30, 45, 47, 49, 50, 51, 52
54. What is the status of CRD?	5, 23, 32, 33, 50, 53
55. What is the status of HPSW?	5, 13, 21, 23, 30, 31, 34, 37, 39, 47, 53
56. Is MSIV leakage control or condenser vacuum maintained?	32

Table 2.1 (Continued)

SUMMARY OF CET EVENTS FOR PEACH BOTTOM

Question Posed by the Event Tree	Dependencies on Prior Questions
57. Is the suppression pool drained before vessel breach?	27, 28, 30, 31, 45, 50, 51, 53
58. What is the level of suppression pool bypass at vessel breach?	20, 26, 43, 44, 49, 51, 53, 57
59. What is level of RB breach/bypass before vessel breach without burn?	5, 20, 23, 26, 32, 47, 49, 52, 53
60. Are the fire sprays actuated before vessel breach without burn?	37, 53, 58, 59
61. Does SGTS fail before vessel breach without burn?	23, 32, 47, 60
62. Does hydrogen burn in RB before vessel breach?	20, 26, 42, 49, 53, 59, 61
63. What is level of RB breach/bypass before vessel breach?	52, 59, 62
64. Are fire systems actuated before vessel breach?	52, 60, 62
65. Does SGTS operate until vessel breach?	61, 62, 64
66. What is containment pressure after containment failure or vent during meltdown?	53
67. Is there injection to the RPV up to the time of vessel breach?	13, 21, 30, 31, 37, 39 48, 53, 54, 55, 57
68. What is the nature of the meltdown?	35, 67
69. What is the mode of vessel breach?	19, 35, 67, 68
70. Is the RPV blown down before significant melt ejection?	38, 48, 53, 68, 69
71. Does direct heating occur?	68, 70
72. Does an ex-vessel steam explosion occur?	68, 69
73. What is pressure rise due to blowdown (effective vapor suppression and reactor cavity dry)?	38, 71
74. What is the pressure rise due to debris quenching?	1, 22, 40, 67
75. Is the suppression pool bypassed or are vents not cleared at vessel breach (effective vapor suppression)	37, 38, 53, 58, 71, 72, 74
76. Does pressurization fail containment at vessel breach?	45, 47, 50, 51, 53, 69, 75
77. What is the base pressure before vessel breach?	45, 47, 50, 53, 69
78. What is the pressure rise at vessel breach?	45, 47, 50, 51, 53, 69, 75
79. Does direct melt-structure attack fail containment at vessel breach?	68, 69, 70, 74
80. What is the location of containment failure after vessel breach?	51, 69, 75, 76, 79
81. For drywell failures, is the failure at the head seal?	52, 79, 80
82. What is the containment leakage level after vessel breach?	53, 69, 75, 76 79, 80, 81
83. Is the suppression pool drained following vessel breach?	57, 76, 80

Table 2.1 (Continued)

SUMMARY OF CET EVENTS FOR PEACH BOTTOM

Question Posed by the Event Tree	Dependencies on Prior Questions
84. Is ac power not available late?	6, 19, 24, 32
85. Is residual heat removal not operating late?	12, 32, 37, 69, 82, 83, 84
86. Do containment sprays not operate following vessel breach?	22, 37, 40, 82, 83, 84
87. Is service water sprayed following vessel breach?	22, 55, 84, 86
88. Is water not supplied to the debris late?	13, 21, 32, 37, 38, 53, 54, 55, 57, 67, 69, 82, 83, 84, 86, 87
89. Is the debris not coolable without water?	69, 71, 72
90. Is the debris coolable with water?	37, 55, 67, 68, 69, 71, 72, 82, 83, 89
91. Is the debris not cooled?	69, 88, 89, 90
92. Is the containment not vented late?	32, 38, 46, 48, 69, 82, 84, 85, 91
93. Does the containment fail late due to temperature?	69, 80, 82, 88, 89, 92
94. Does the containment fail late?	69, 82, 85, 88, 91, 92, 93
95. What is the location of late containment leakage?	58, 80, 82, 88, 92, 93, 94
96. For drywell failures, is the failure at the head seal?	81, 88, 94, 95
97. What is the level of late containment leakage?	82, 88, 92, 94, 95, 96
98. Is the suppression pool drained late?	83, 94, 95, 97
99. What is the level of late suppression pool bypass?	20, 25, 26, 38, 41, 49, 58, 69, 95, 98
100. Do drywell sprays continue?	37, 55, 86, 87, 94, 98
101. What is the level of late RB bypass without a burn?	59, 63, 69, 79, 96, 97
102. Are fire systems operating late without a late hydrogen burn?	37, 64, 96, 97, 99, 101
103. Does standby gas treatment not work late without late hydrogen burn?	65, 84, 102
104. Does hydrogen burn in the RB after vessel breach?	91, 97, 101, 103
105. What is the level of late RB bypass?	97, 101, 103, 104
106. Are fire systems operating late?	96, 102, 104
107. Does standby gas treatment work late?	103, 106

Table 2.2

NOMENCLATURE USED IN PLANT DAMAGE STATE NAMES

Character	Interpretation
B	Failure of offsite and onsite ac power (station blackout)
C	Failure to scram (control blades remain withdrawn)
P	Failure of at least one SRV to reclose
R	Failure to successfully vent containment
S	Failure of standby liquid control (no boron injection)
T	Damage state initiated by a plant transient
U	Failure of high-pressure injection to core
V	Failure of low-pressure injection to core
X	Failure to depressurize the reactor vessel
2	Leakage from containment insufficient to relieve internal pressure
23	Rupture of containment resulting in pressure relief

Table 2.3

MEAN FREQUENCY OF PLANT DAMAGE STATES [9]

Plant Damage State	Mean Frequency (yr^{-1})
TBUX	4.2×10^{-6}
TB	2.3×10^{-6}
TCUX	4.8×10^{-7}
TCSRX2	2.4×10^{-7}
TBUXP (Subset of TBUX, combined with TBUP in [1])	2.0×10^{-7}
TCSX	1.7×10^{-7}
TBU	1.6×10^{-7}
TBP (Subset of TB)	1.2×10^{-7}
TCSRV23	2.5×10^{-8}
TBUP (Subset of TBU)	8×10^{-9}
Total (96% of total core damage frequency)	7.9×10^{-6}

Table 3.1

Central Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
1.	Subcooled pool, containment failure at vessel breach, late DW rupture, RB bypass.	--	0.062	0.115	--	--	--	--
2.	Subcooled pool, containment failure at vessel breach, late DW rupture.	--	0.222	0.407	--	--	--	--
3.	Subcooled pool, late DW rupture, RB bypass.	--	0.017	0.012	--	--	--	--
4.	Subcooled pool, late DW leak, RB bypass.	--	0.076	0.065	--	--	--	--
5.	Subcooled pool, late DW rupture.	--	0.057	0.041	--	--	--	--
6.	Subcooled pool, late DW leak.	--	0.057	0.042	--	--	--	--
7.	Subcooled pool, late WW rupture/vent, RB bypass.	--	0.022	0.017	--	--	--	--
8.	Subcooled pool, some CCI, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	0.011	--	--	--	--	--
9.	Subcooled pool, some CCI, containment failure at vessel breach, late DW rupture, drywell spray.	--	0.038	0.023	--	--	--	--
10.	Subcooled pool, some CCI, late WW rupture/vent, drywell spray.	--	0.037	0.012	--	--	--	--
11.	Subcooled pool, no CCI, containment failure at vessel breach, late DW rupture, drywell spray.	--	0.038	0.029	--	--	--	--
12.	Subcooled pool, no CCI, late WW rupture/vent, drywell spray.	--	--	0.011	--	--	--	--
13.	Subcooled pool, no CCI, containment intact, drywell spray.	--	0.036	--	--	--	--	--
14.	Subcooled pool, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	0.013	--	--	--	--	--
15.	Subcooled pool, containment failure at vessel breach, late DW rupture, drywell spray.	--	0.045	0.026	--	--	--	--

Table 3.1

Central Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
16.	Subcooled pool, late WW rupture/vent, drywell spray.	--	0.055	0.031	--	--	--	--
17.	Subcooled pool, no CCI, containment failure at vessel breach, late DW rupture, drywell spray.	--	0.018	0.010	--	--	--	--
18.	Subcooled pool, no CCI, containment intact, drywell spray.	--	0.023	0.013	--	--	--	--
19.	Subcooled pool, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	--	--	0.062	--	--	--
20.	Subcooled pool, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	0.238	--	--	--
21.	Subcooled pool, late WW rupture/vent, drywell spray.	--	--	--	0.129	--	--	--
22.	Subcooled pool, no CCI, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	0.022	--	--	--
23.	Subcooled pool, no CCI, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	0.208	--	--	--
24.	Subcooled pool, no CCI, late WW rupture/vent, drywell spray.	--	--	--	0.032	--	--	--
25.	Subcooled pool, no CCI, containment intact, drywell spray.	--	--	--	0.040	--	--	--
26.	Subcooled pool, no vessel breach, no CCI, containment intact, drywell spray.	--	0.126	0.073	--	--	--	--
27.	Subcooled pool, no vessel breach, no CCI, containment intact, drywell spray.	--	0.014	--	0.224	--	--	--
28.	Saturated pool, containment failure at vessel breach, late DW rupture, RB bypass.	--	--	--	--	--	--	--
29.	Saturated pool, containment failure at vessel breach, late DW rupture.	0.268	--	--	--	--	--	--

Table 3.1

Central Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
30.	Saturated pool, late DW leak, RB bypass.	0.034	--	--	--	--	--	--
31.	Saturated pool, early WW rupture/vent, late DW rupture, drywell spray.	0.012	--	--	--	--	--	--
32.	Saturated pool, early WW rupture/vent, late DW rupture.	0.012	--	--	--	--	--	--
33.	Saturated pool, early WW rupture/vent, drywell spray.	0.010	--	--	--	--	--	--
34.	Saturated pool, containment failure at vessel breach, late DW rupture, drywell spray.	0.016	--	--	--	--	--	--
35.	Saturated pool, containment failure at vessel breach, late DW rupture.	0.014	--	--	--	--	--	--
36.	Saturated pool, late WW rupture/vent, drywell spray.	0.036	--	--	--	--	--	--
37.	Saturated pool, no CCI, early WW rupture/vent, drywell spray.	0.012	--	--	--	--	--	--
38.	Saturated pool, no CCI, early WW rupture/vent.	0.012	--	--	--	--	--	--
39.	Saturated pool, no CCI, containment failure at vessel breach, late DW rupture, drywell spray.	0.012	--	--	--	--	--	--
40.	Saturated pool, no CCI, containment failure at vessel breach, late DW rupture.	0.010	--	--	--	--	--	--
41.	Saturated pool, no CCI, containment intact, drywell spray.	0.019	--	--	--	--	--	--
42.	Saturated pool, early DW rupture.	--	--	--	--	0.011	--	--
43.	Saturated pool, early DW rupture, RB bypass, drywell spray.	--	--	--	--	--	--	0.034
44.	Saturated pool, early DW rupture, RB bypass, late DW rupture, RB bypass.	--	--	--	--	--	--	0.032
45.	Saturated pool, early DW head leak, containment failure at vessel breach, late DW rupture, RB	--	--	--	--	--	0.013	--

Table 3.1

Central Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
	bypass, drywell spray.							
46.	Saturated pool, early DW head leak, containment failure at vessel breach, late DW rupture, RB bypass.	--	--	--	--	--	0.011	--
47.	Saturated pool, early DW head leak, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	--	--	0.044	--
48.	Saturated pool, early DW head leak, containment failure at vessel breach, late DW rupture.	--	--	--	--	--	0.040	--
49.	Saturated pool, early DW head leak, late DW leak, RB bypass, drywell spray.	--	--	--	--	--	0.014	--
50.	Saturated pool, early DW head leak, late DW leak, RB bypass.	--	--	--	--	--	0.012	--
51.	Saturated pool, early DW head leak, late DW head leak, drywell spray.	--	--	--	--	--	0.016	--
52.	Saturated pool, early DW head leak, late DW head leak.	--	--	--	--	--	0.015	--
53.	Saturated pool, early DW rupture, late DW rupture, RB bypass, drywell spray.	--	--	--	--	--	--	0.027
54.	Saturated pool, early DW rupture, late DW rupture, RB bypass.	--	--	--	--	--	--	0.025
55.	Saturated pool, early DW rupture, late DW rupture, drywell spray.	--	--	--	--	--	--	0.095
56.	Saturated pool, early DW rupture, late DW rupture.	--	--	--	--	--	--	0.089
57.	Saturated pool, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	--	--	--	--	0.016	--	--
58.	Saturated pool, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	--	--	--	--	--	0.016	--

Table 3.1

Central Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
59.	Saturated pool, early WW rupture/vent, RB bypass, late DW rupture, RB bypass.	--	--	--	--	0.014	--	--
60.	Saturated pool, early WW rupture/vent, RB bypass, late DW rupture, RB bypass.	--	--	--	--	--	0.015	--
61.	Saturated pool, early WW rupture/vent, RB bypass, drywell spray.	--	--	--	--	0.010	--	--
62.	Saturated pool, early WW rupture/vent, late DW rupture, RB bypass, drywell spray.	--	--	--	--	0.012	0.013	--
63.	Saturated pool, early WW rupture/vent, late DW rupture, RB bypass.	--	--	--	--	0.011	0.011	--
64.	Saturated pool, early WW rupture/vent, late DW rupture, drywell spray.	--	--	--	--	0.044	0.046	0.018
65.	Saturated pool, early WW rupture/vent, late DW rupture.	--	--	--	--	0.040	0.041	0.017
66.	Saturated pool, early WW rupture/vent, drywell spray.	--	--	--	--	0.028	0.023	--
67.	Saturated pool, early WW rupture/vent.	--	--	--	--	0.026	0.021	--
68.	Saturated pool, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	--	--	--	0.010	--	--
69.	Saturated pool, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	--	0.036	--	--
70.	Saturated pool, containment failure at vessel breach, late DW rupture.	--	--	--	--	0.033	--	--
71.	Saturated pool, late WW rupture/vent, RB bypass, drywell spray.	--	--	--	--	0.011	--	--
72.	Saturated pool, late WW rupture/vent, drywell spray.	--	--	--	--	0.038	--	--
73.	Saturated pool, no CCI, early DW rupture, RB bypass, drywell spray.	--	--	--	--	--	--	0.028

Table 3.1

Central Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
74.	Saturated pool, no CCI, early DW rupture, RB bypass, late DW rupture, RB bypass.	--	--	--	--	--	--	0.026
75.	Saturated pool, no CCI, early DW head leak, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	--	--	0.056	--
76.	Saturated pool, no CCI, early DW head leak, containment failure at vessel breach, late DW rupture.	--	--	--	--	--	0.050	--
77.	Saturated pool, no CCI, early DW head leak, drywell spray.	--	--	--	--	--	0.030	--
78.	Saturated pool, no CCI, early DW head leak.	--	--	--	--	--	0.027	--
79.	Saturated pool, no CCI, early DW rupture, drywell spray.	--	--	--	--	--	0.011	0.099
80.	Saturated pool, no CCI, early DW rupture.	--	--	--	--	--	--	0.093
81.	Saturated pool, no CCI, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	--	--	--	--	0.014	--	
82.	Saturated pool, no CCI, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	--	--	--	--	--	0.016	--
83.	Saturated pool, no CCI, early WW rupture/vent, RB bypass, late DW rupture, RB bypass.	--	--	--	--	0.013	0.014	--
84.	Saturated pool, no CCI, early WW rupture/vent, RB bypass, drywell spray.	--	--	--	--	0.010	--	--
85.	Saturated pool, no CCI, early WW rupture/vent, late DW rupture, drywell spray.	--	--	--	--	0.049	0.057	0.016
86.	Saturated pool, no CCI, early WW rupture/vent, late DW rupture.	--	--	--	--	0.045	0.051	0.015

Table 3.1

Central Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
87.	Saturated pool, no CCI, early WW rupture/vent, drywell spray.	--	--	--	--	0.036	0.030	0.012
88.	Saturated pool, no CCI, early WW rupture/vent.	--	--	--	--	0.033	0.027	0.011
89.	Saturated pool, no CCI, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	--	0.035	--	--
90.	Saturated pool, no CCI, containment failure at vessel breach, late DW rupture.	--	--	--	--	0.032	--	--
91.	Saturated pool, no CCI, containment intact, drywell spray.	--	--	--	--	0.025	--	--
92.	Saturated pool, no vessel breach, no CCI, early WW rupture/vent, RB bypass, drywell spray.	0.011	--	--	--	--	--	--
93.	Saturated pool, no vessel breach, no CCI, early WW rupture/vent, RB bypass.	0.011	--	--	--	--	--	--
94.	Saturated pool, no vessel breach, no CCI, early WW rupture/vent, late WW rupture/vent, drywell spray.	0.040	--	--	--	--	--	--
95.	Saturated pool, no vessel breach, no CCI, early WW rupture/vent, late WW rupture/vent.	0.040	--	--	--	--	--	--
96.	Saturated pool, no vessel breach, no CCI, containment intact, drywell spray.	0.106	--	--	--	--	--	--
97.	Saturated pool, no vessel breach, no CCI, containment intact, drywell spray.	0.012	--	--	--	--	--	--
98.	Saturated pool, no vessel breach, no CCI, early DW rupture, RB bypass, drywell spray.	--	--	--	--	--	--	0.018
99.	Saturated pool, no vessel breach, no CCI, early DW rupture, RB bypass.	--	--	--	--	--	--	0.018
100.	Saturated pool, no vessel breach, no CCI, early DW	--	--	--	--	--	--	0.064

Table 3.1

Central Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
	rupture, drywell spray.							
101.	Saturated pool, no vessel breach, no CCI, early DW rupture.	--	--	--	--	--	--	0.064
102.	Saturated pool, no vessel breach, no CCI, early WW rupture/vent, drywell spray.	--	--	--	--	0.033	--	0.023
103.	Saturated pool, no vessel breach, no CCI, early WW rupture/vent.	--	--	--	--	0.033	--	0.023
104.	Saturated pool, no vessel breach, no CCI, containment intact, drywell spray.	--	--	--	--	0.138	--	--
	TOTALS:	0.755	0.967	0.928	0.955	0.827	0.721	0.858

Table 3.2

Optimistic Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
5.	Subcooled pool, late DW rupture.	--	--	0.011	--	--	--	--
6.	Subcooled pool, late DW leak.	--	0.087	0.119	--	--	--	--
7.	Subcooled pool, late WW rupture/vent, RB bypass.	--	0.162	0.222	--	--	--	--
105.	Subcooled pool, late WW leak.	--	0.052	0.072	--	--	--	--
12.	Subcooled pool, no CCI, late WW rupture/vent, drywell spray.	--	--	0.172	--	--	--	--
18.	Subcooled pool, no CCI, containment intact, drywell spray.	--	0.247	0.033	--	--	--	--
24.	Subcooled pool, no CCI, late WW rupture/vent, drywell spray.	--	--	--	0.100	--	--	--
25.	Subcooled pool, no CCI, containment intact, drywell spray.	--	--	--	0.081	--	--	--
26.	Subcooled pool, no vessel breach, no CCI, containment intact, drywell spray.	--	0.420	0.333	0.810	--	--	--
106.	Saturated pool, late DW leak.	0.034	--	--	--	--	--	--
107.	Saturated pool, late WW rupture/vent, RB bypass.	0.064	--	--	--	--	--	--
108.	Saturated pool, late WW leak.	0.021	--	--	--	--	--	--
37.	Saturated pool, no CCI, late WW rupture/vent, drywell spray.	0.038	--	--	--	--	--	--
41.	Saturated pool, no CCI, containment intact, drywell spray.	0.074	--	--	--	--	--	--
109.	Saturated pool, no CCI, early DW rupture, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	--	--	0.020	--
87.	Saturated pool, no CCI, early WW rupture/vent, drywell spray.	--	--	--	--	--	0.871	0.080

Table 3.2

Optimistic Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
88.	Saturated pool, no CCI, early WW rupture/vent.	--	--	--	--	--	0.093	--
	Saturated pool, no CCI, late WW rupture/vent, drywell spray.	--	--	--	--	0.100	--	--
110.	Saturated pool, no CCI, containment intact, drywell spray.	--	--	--	--	0.081	--	--
96.	Saturated pool, no vessel breach, no CCI, containment intact, drywell spray.	0.744	--	--	--	--	--	--
100.	Saturated pool, no vessel breach, no CCI, early DW rupture, drywell spray.	--	--	--	--	--	--	0.081
102.	Saturated pool, no vessel breach, no CCI, early WW rupture/vent, drywell spray.	--	--	--	--	--	--	0.729
103.	Saturated pool, no vessel breach, no CCI, early WW rupture/vent.	--	--	--	--	--	--	0.081
104.	Saturated pool, no vessel breach, no CCI, containment intact, drywell spray.	--	--	--	--	0.810	--	--
	TOTALS:	0.975	0.969	0.962	0.991	0.991	0.985	0.971

Table 3.3

Pessimistic Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
1.	Subcooled pool, containment failure at vessel breach, late DW rupture, RB bypass.	--	0.717	0.959	--	--	--	--
8.	Subcooled pool, some CCI, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	0.084	--	--	--	--	--
111.	Subcooled pool, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	--	0.147	0.030	--	--	--	--
19.	Subcooled pool, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	--	--	0.888	--	--	--
22.	Subcooled pool, no CCI, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	0.090	--	--	--
28.	Saturated pool, containment failure at vessel breach, late DW rupture, RB bypass.	0.572	--	--	--	--	--	--
112.	Saturated pool, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	0.332	--	--	--	--	--	--
113.	Saturated pool, no CCI, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	0.020	--	--	--	--	--	--
43.	Saturated pool, early DW rupture, RB bypass, drywell spray.	--	--	--	--	--	--	0.926
114.	Saturated pool, early DW leak, RB bypass, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	--	--	--	0.024	--	--
115.	Saturated pool, early DW head leak, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	--	--	--	0.024	--	--
116.	Saturated pool, early WW rupture/vent, RB bypass,	--	--	--	--	--	0.218	--

Table 3.3

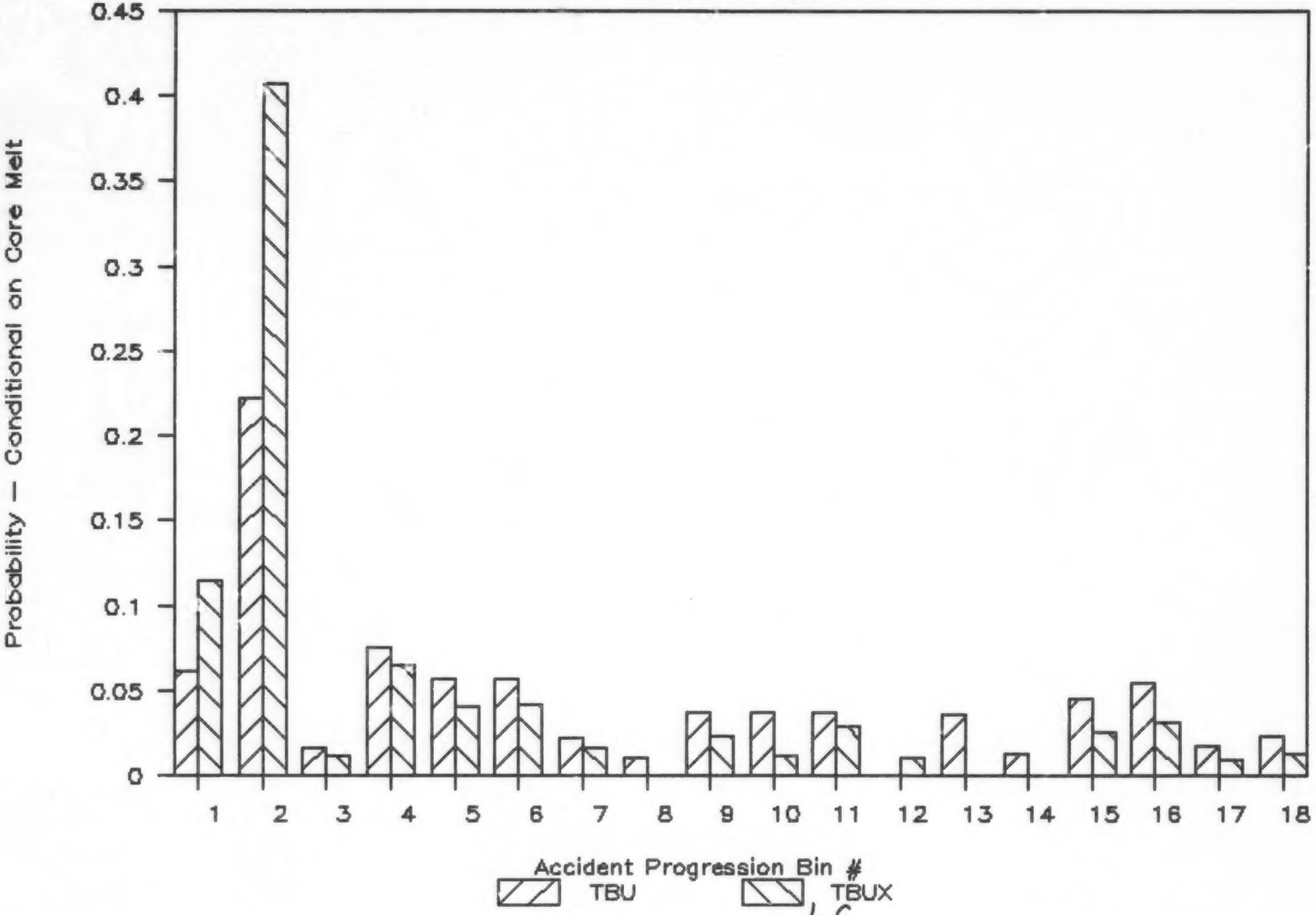
Pessimistic Evaluation of Peach Bottom Containment Performance

BIN #	PROGRESSION BIN	TB	TBU	TBUX	TCUX	TCSX	TCSRX2	TCSRV23
	containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.							
57.	Saturated pool, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	--	--	--	--	--	0.682	--
68.	Saturated pool, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	--	--	--	0.832	--	--
73.	Saturated pool, no CCI, early DW rupture, RB bypass, drywell spray.	--	--	--	--	--	--	0.064
117.	Saturated pool, no CCI, early WW rupture/vent, RB bypass, containment failure at vessel breach, late DW rupture, RB bypass, drywell spray.	--	--	--	--	--	0.024	--
81.	Saturated pool, no CCI, early WW rupture/vent, RB bypass, late DW rupture, RB bypass, drywell spray.	--	--	--	--	--	0.076	--
89.	Saturated pool, no CCI, containment failure at vessel breach, late DW rupture, drywell spray.	--	--	--	--	0.084	--	--
	TOTALS:	0.925	0.947	0.989	0.978	0.965	1.000	0.990

Figure 3.1

Dominant Accident Progression Bins

Short-Term Station Blackout



46

Figure 3.2

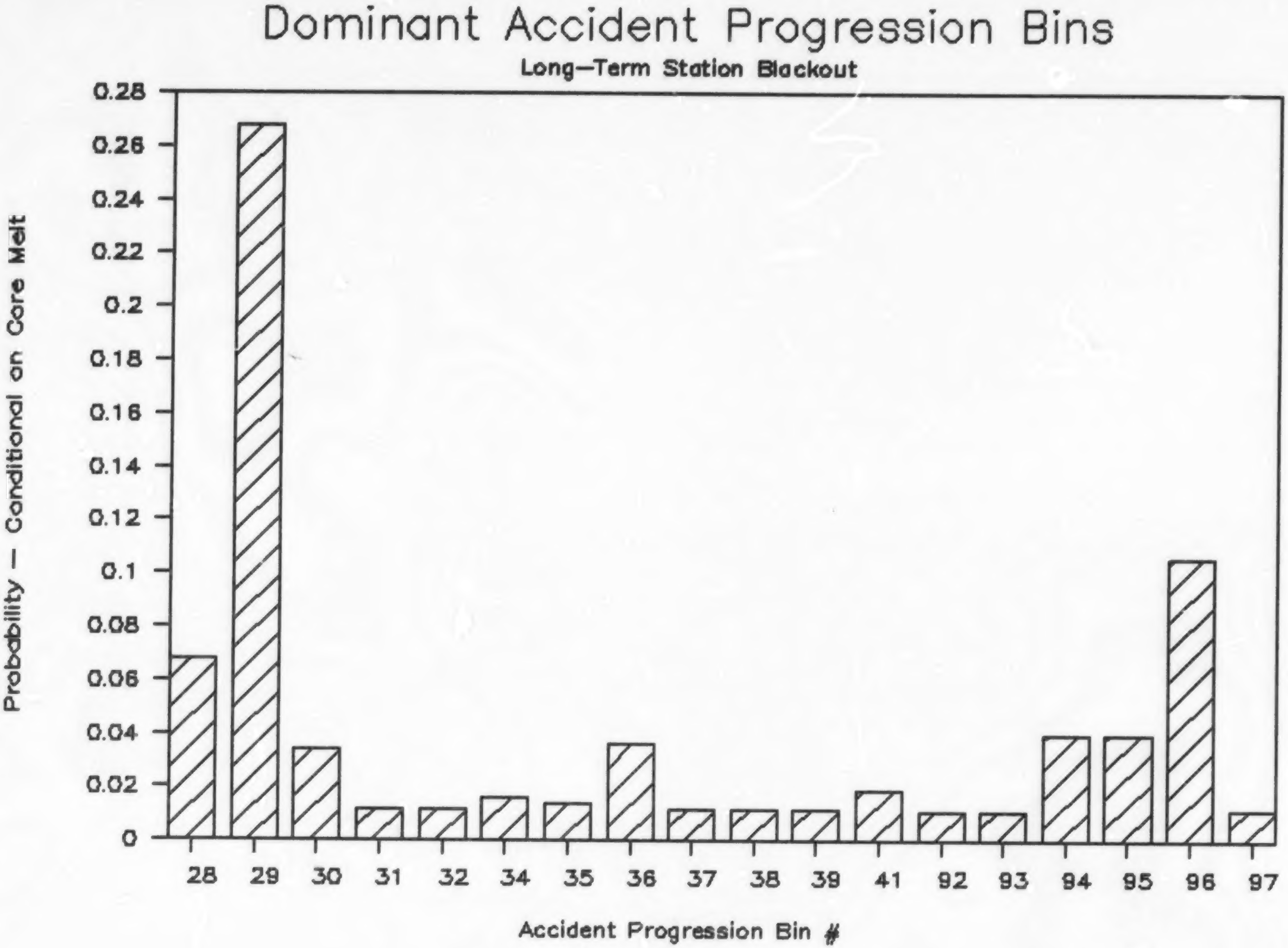

Dominant Accident Progression Bins
Long–Term Station Blackout
Probability — Conditional on Core Melt
0.28
0.26
0.24
0.22
0.2
0.18
0.16
0.14
0.12
0.1
0.08
0.06
0.04
0.02
0
28
29
30
31
32
34
35
36
37
38
39
41
92
93
94
95
96
97
Accident Progression Bin #

Figure 3.3

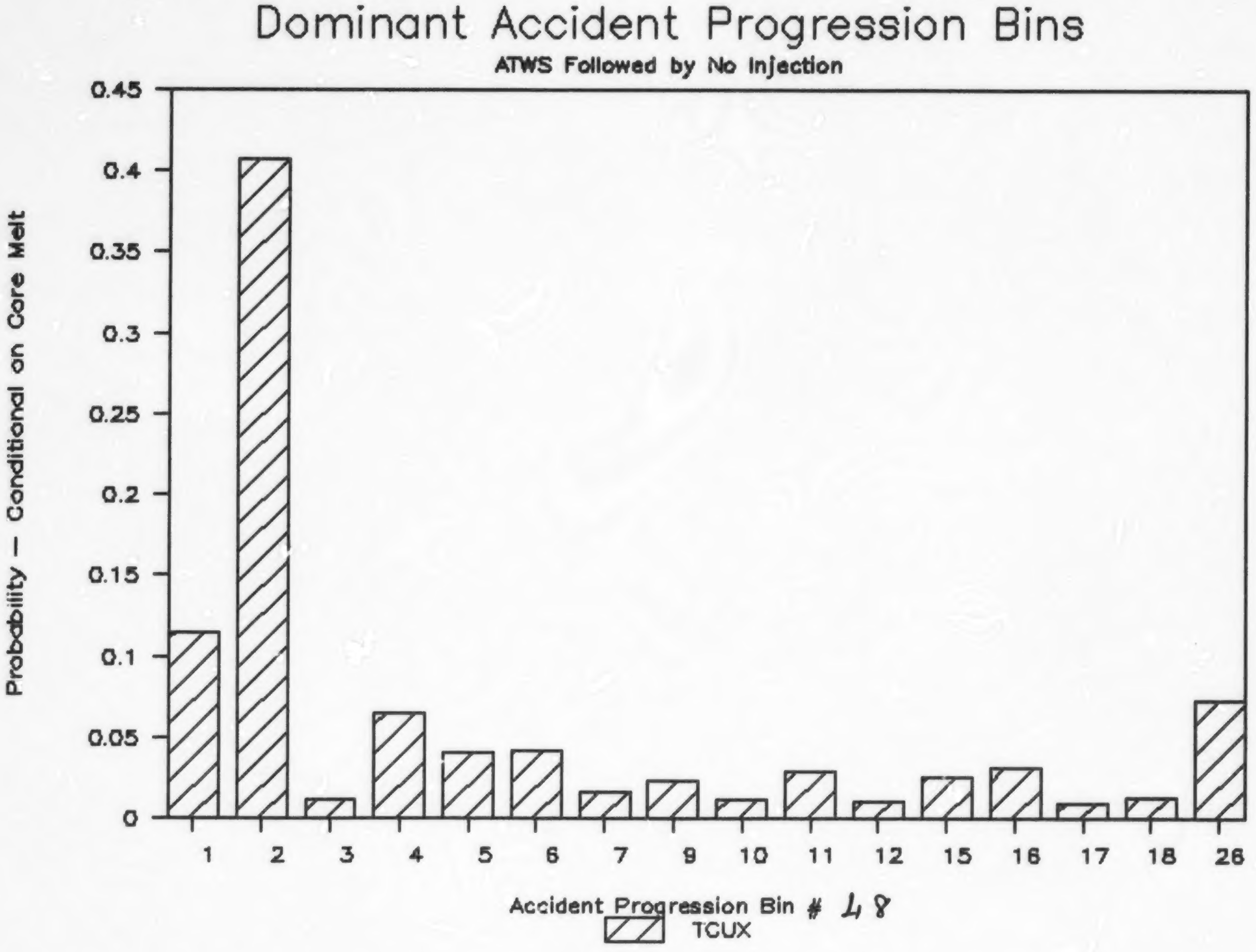
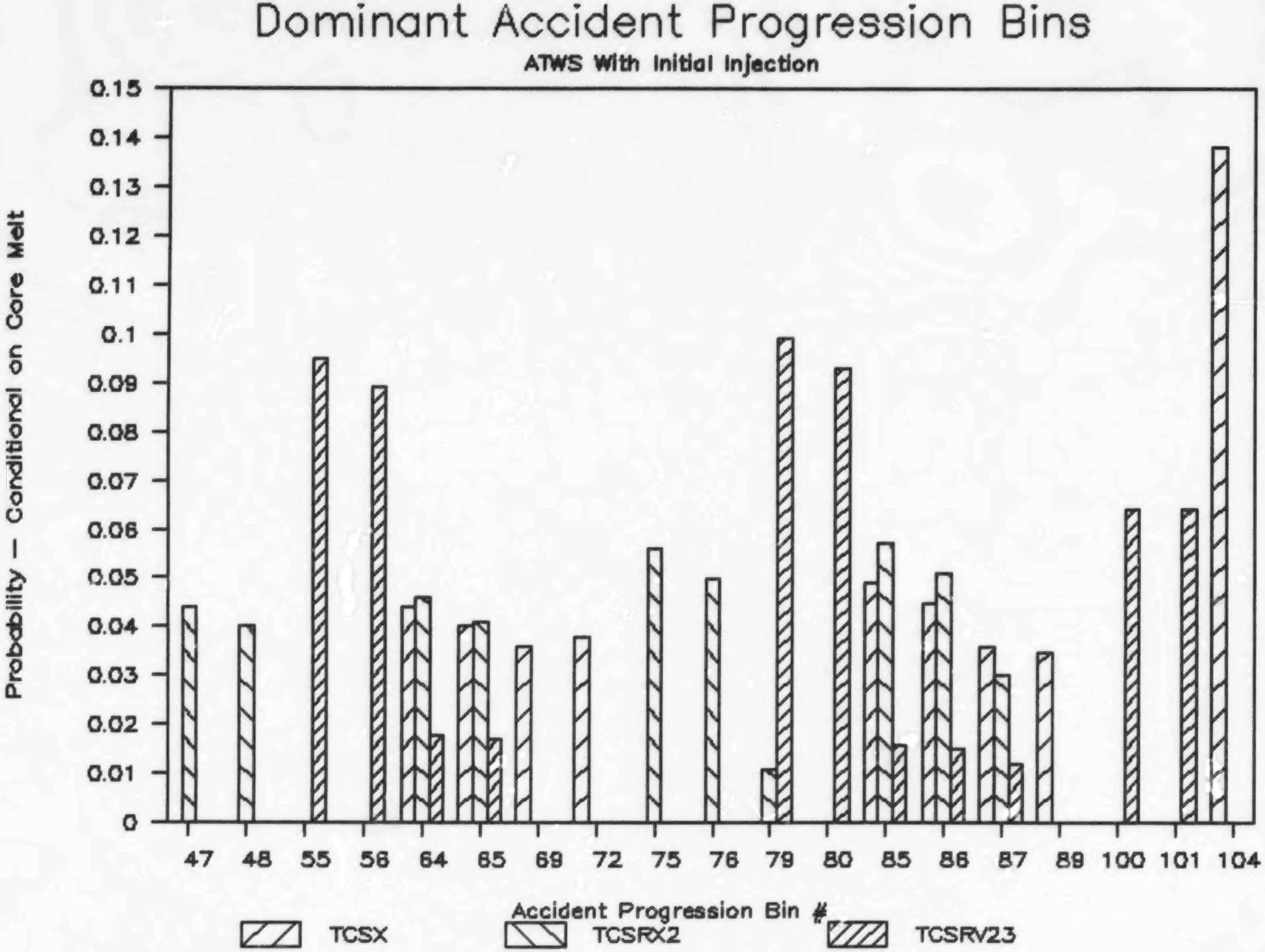

Dominant Accident Progression Bins
ATWS Followed by No Injection
Probability — Conditional on Core Melt
0.45
0.4
0.35
0.3
0.25
0.2
0.15
0.1
0.05
0
1
2
3
4
5
6
7
9
10
11
12
15
16
17
18
26
Accident Progression Bin # 48
TCUX

Figure 3.4


Dominant Accident Progression Bins
ATWS With Initial Injection
Probability — Conditional on Core Melt
0.15
0.14
0.13
0.12
0.11
0.1
0.09
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
47
48
55
56
64
65
69
72
75
76
79
80
85
86
87
89
100
101
104
Accident Progression Bin #
TCSX
TCSRX2
TCSRV23

5. REFERENCES

1. (SARRP) C. N. Amos, et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Peach Bottom, Unit 1," NUREG/CR-4551, Volume 3 (Draft Report for Comment), Sandia National Laboratories, February 1987.

2. (1150) "Reactor Risk Reference Report," NUREG-1150 (Draft Report for Comment), U.S. Nuclear Regulatory Commission, February 1987.

3. (0956) "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956, U.S. Nuclear Regulatory Commission, August 1985.

4. (BMI) J. A. Gieseke, et al., "Radionuclide Release Under Specific Accident Conditions," BMI-2104, Battelle Columbus Laboratories, July 1984.

5. (SARRP) R. S. Denning, et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios: BWR, Mark I Design," Vol. 1, NUREG/CR-4624 Battelle Columbus Laboratories, July 1986.

6. (SARRP) M. L. Corradini, et al., "A Review of the SARRP Containment Event Trees," NUREG/CR-4569, University of Wisconsin, May 1986.

7. (SARRP) M. L. Corradini, "Additional Comments on the Peach Bottom and Grand Gulf Containment Event Trees," letter report to U.S. Nuclear Regulatory Commission, October 13, 1986.

8. (SARRP) "Comments on Peach Bottom Containment Event Tree," letter from V. Boyer, Philadelphia Electric Company, to B. Morris, U.S. Nuclear Regulatory Commission, September 19, 1986.

9. (ASEP) A. M. Kolaczkowski, et al., "Reference Plant Accident Sequence Likelihood Characterization: Peach Bottom, Unit 2," NUREG/CR-4550, Vol. 3, May 1986 (DRAFT).

10. (RSS) "Reactor Safety Study," WASH-1400, NUREG-75/014, U.S. Nuclear Regulatory Commission, October 1975.

11. (CLWG) "Estimates of Early Containment Loads from Core Melt Accidents, Report of the Containment Loads Working Group," NUREG-1079 (Draft for Comment), U.S. Nuclear Regulatory Commission, December 1985.

12. (CLWG) Consensus Summaries for Standard Problems 1 through 6. Letter Reports Addressed to J. Telford, U.S. Nuclear Regulatory Commission, May-June 1984.

13. (CPWG) Containment Performance Working Group, "Containment Leak Rate Estimates," NUREG-1037 (Preliminary Draft), U.S. Nuclear Regulatory Commission, 1984.

14. (QUEST) R.J. Lipinski, et al., "Uncertainty in Radionuclide Release Under Specific LWR Accident Conditions," SAND84-0410, Sandia National Laboratories, February 1985.

15. (IDCOR) IDCOR Program Reports, IDCOR Tasks 21.1 (Risk Reduction Potential) and 23.1 (Integrated Containment Analysis), Technology for Energy Corporation, November 1984.

16. (SASA) D. H. Cook, et al., "Loss of Decay Heat Removal Sequences at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-2973, ORNL/TM-8532, Oak Ridge National Laboratory, May 1983.

17. (SASA) R. M. Harrington, et al., "ATWS at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-3470, ORNL/TM-8902, Oak Ridge National Laboratory, July 1984.

18. (SAUNA) J. B. Rivard, et al., "Identification of Severe Accident Uncertainties," NUREG/CR-3440, SAND83-1689, Sandia National Laboratories, September 1984.

19. (ASEP) A. M. Kolaczkowski, et al., "Interim Report on Accident Sequence Likelihood Reassessment," Draft Report, Sandia National Laboratories, August 1983.

20. (PATF) A. S. Benjamin, et al., "SARRP - Risk Rebaselining and Risk Reduction Analysis," *11th Water Reactor Safety Research Information Meeting*, NUREG/CR-0048, Vol.3, January 1984.

21. (SERG) Steam Explosion Review Group, "A Review of the Current Understanding of the Potential for Containment Failure Arising from In-Vessel Steam Explosions," NUREG-1116, U.S. Nuclear Regulatory Commission, February 1985.

22. (HIPS) High Pressure Ejection Test Series, Sandia National Laboratories, 1985.

23. (RSSMAP) "Reactor Safety Study, Methodology Applications Program," NUREG/CR-1659, SAND80-1897, Volumes 1-4, Sandia National Laboratories, 1981-1982.

24. (GESSAR) "GESSAR-II, BWR/6 Standard Plant Probabilistic Risk Assessment," General Electric Co., 1982.

25. (FSAR) Final Safety Analysis Report for Peach Bottom Unit 2.

26 (AMES) L. Greimann, et al., "Final Report, Containment Analysis Techniques, A State-of-the-Art Summary," NUREG/CR-3653, SAND83-7463, Ames Laboratory and Sandia National Laboratories, March 1984.

27. (NRR) M. B. Weinstein, "Primary Containment Leakage Integrity: Availability and Review of Failure Experience," Nuclear Safety, 21, 1980.

28. (NRR) W. G. Lyons, Presentation at NRC/IDCOR Meeting on Containment Loads and Fission Product Behavior, U.S. Nuclear Regulatory Commission, May 15-17, 1984.

29. (HYDR) A L. Camp, et al., "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, SAND82-1137, Sandia National Laboratories, August, 1983.

30 (LPS) S. J. Niemczyk, et al., "The Consequences from Liquid Pathways After a Meltdown Accident," NUREG/CR-1596, SAND80-1669, Sandia National Laboratories, June 1981.

31 (EPRI) "Loss of Offsite Power at Nuclear Power Plants: Data and Analysis," EPRI NP-2301, Interim Report, Electric Power Research Institute, March 1982.

APPENDIX A

Detailed Results of the Peach Bottom Containment Event Tree Analysis

APPENDIX A

Detailed Results of the Containment Event Tree Analysis

The tables in this appendix present the detailed central estimate results of the Peach Bottom containment event tree analysis. The results for each plant damage state are presented in a separate table:

TB: Table A-2
TBU: Table A-3
TBUX: Table A-4
TCUX: Table A-5
TCSX: Table A-6
TCSRX2: Table A-7
TCSRV23: Table A-8

Each table is essentially a listing of the output from the EVNTRE computer code for the run made to analyze that particular plant damage state.

The tables provide the probability (conditional on the occurrence of the plant damage state) of each accident progression bin which was calculated to have a probability greater than 1×10^{-5}. Accident progression bins for each plant damage state are listed in order of decreasing probability. Each bin is defined in terms of ten "dimensions." The ten dimensions, or aspects of the accident progression, considered in binning the pathways through the CET each consider different aspects of the pathway which may impact containment failure mode and/or source term. An accident progression bin is defined by a unique combination of outcomes for each of the ten dimensions. Stated another way, the "attributes" of the accident progression bin are defined in terms of the answers to ten questions asked about the progression of the accident. The questions asked about the pathway taken in the CET for a particular accident progression (i.e. the dimensions of the bin) are as follows:

1. What is the general scenario (eg. station blackout, station blackout with power recovery, ATWS)?

2. What (if anything) caused the containment to rupture?

3. What is the location of containment leakage before vessel breach? Immediately after vessel breach? Late in the accident?

4. How large was the leakage from containment before vessel breach? Immediately after vessel breach? Late in the accident?

5. What were the conditions in the RPV during core damage?

54

BEST COPY AVAILABLE

6. How did the RPV fail?

7. To what extent was the suppression pool bypassed?

8. To what extent was the reactor building (and the standby gas treatment system) bypassed?

9. Under what conditions did core-concrete interaction occur?

10. When did the drywell sprays operate?

Table A-1 lists the possible outcomes for each of the dimensions of the accident progression bins along with the abbreviation of that outcome used in Tables A-2 through A-8. Using Table A-1 as a key, the attributes of all the accident progression bins realized in the central estimate results can be understood.

Table A-1
Key to Nomenclature Used in Detailed Results Presentations

Scenario:

TB-S-ER = Station blackout, long-term sequence, early recovery
TB-S-LR = as above except late recovery
TB-S-nR = as above except no recovery
TB-F-ER = Station blackout, short-term sequence, early recovery
TB-F-LR = as above except late recovery
TB-F-nR = as above except no recovery
TC-V = ATWS with core vulnerable due to possible containment failure affecting continuation of core cooling
TC-F = ATWS, fast melt sequence
TC-S = ATWS, slow melt sequence

Containment Failure Mode (rupture only)

CFD-ID = Steam explosion
CFD-IP = Overpressure failure at vessel breach
CFD-E4P = Overpressure during core degradation
CFD-E4D = Suppression pool lift induced failure
CFD-E4V = Venting during core degradation
CFD-E3P = Overpressure before core melt
CFD-E3V = Venting before core melt
CFD-IM = Drywell wall meltthrough at vessel breach
CFD-LP = Late overpressure failure
CFD-LT = Late thermal failure
CRD-LV = Late venting
CFD-n = No rupture (leak possible)

Containment Failure (leak or rupture) Location

CLC-xyz x = early; y = at vessel breach; z = late
n = None
D = Drywell
W = Wetwell

Containment Failure Level

CLV-xyz x = early; y = at vessel breach; z = late
0 = Insignificant
2 = A leak but insufficient to depressurize containment
3 = Rupture

In-Vessel Progression (of core damage)

CD-nIHH = No injection, high vessel pressure, high amounts of hydrogen
CD-nIHM = as above except moderate amount of hydrogen
CD-nIHL = as above except low amounts of hydrogen
CD-nILH = No injection, low vessel pressure, high amounts of hydrogen
CD-nILM = as above except moderate amounts of hydrogen

Table A.1 (Continued)

In-Vessel Progression (Continued)

CD-nILL = as above except low amounts of hydrogen
CD-IHH = Some injection; high vessel pressure, high amounts of hydrogen
CD-IHM = as above except moderate amounts of hydrogen
CD-IHL = as above except low amounts of hydrogen
CD-ILH = Some injection, low vessel pressure, high amount of hydrogen
CD-ILM = as above except moderate amounts of hydrogen
CD-ILL = as above except low amounts of hydrogen

Vessel Breach Mode

VBD-nB = No vessel breach
VBD-F = Flow type melt
VBD-F-B = Flow type melt with suppression pool bypass at vessel breach
VBD-DH = Direct heating
VBD-DH-B = Direct heating with suppression pool bypass at vessel breach
VBD-SE = Ex-vessel steam explosion
VBD-SE-B = Ex-vessel steam explosion with suppression pool bypass at vessel breach
VBD-S = Slump melt
VBD-S-B = Slump melt and suppression pool bypass at vessel breach

Suppression Pool Bypass

SPB-xy x = early; y = late
n = No
pv = S/RV vacuum breaker partially stuck open
v = S/RV vacuum breaker stuck open
y = Yes

Reactor Building Bypass

RBB-xy x = early; y = late
n = Through standby gas treatment system (no bypass)
2 = Blowout panels
3 = Path through refueling bay
4 = Large failure

Core-Concrete Interaction

nCCI = No interaction
WCCI = Wet interaction
DCCI = Dry interaction

Table A-1 (Continued)

Core-Concrete Interaction (Continued)

nCCI-LR = Dry interaction quenched late
WCCI-LR = Wet interaction quenched late

Drywell Sprays

nSPY = No sprays at anytime
SPY-E = Sprays until containment failure then stop
SPY-L = Sprays start after vessel breach
SPY-EL = Sprays throughout accident

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
2.026E-01	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
8.612E-02	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
5.706E-02	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
3.323E-02	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-22	nCCI	SRY-EL
3.259E-02	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-22	nCCI	SPY-E
2.788E-02	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
2.725E-02	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
2.608E-02	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	DCCI	nSPY
1.875E-02	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.543E-02	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
1.476E-02	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
9.920E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	nCCI	SRY-EL
9.805E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	nCCI	SPY-E
9.734E-03	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
9.500E-03	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
9.326E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-44	nCCI	SRY-EL
9.174E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-44	nCCI	SPY-E
8.564E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
7.838E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	WCCI	SRY-EL
7.749E-03	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n4	WCCI	SRY-EL
7.681E-03	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
7.639E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	WCCI	SPY-E
7.624E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
7.388E-03	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	DCCI	nSPY
7.199E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-22	nCCI	SRY-EL
7.071E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-22	nCCI	SPY-E
6.792E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
6.595E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	nCCI	SRY-EL
6.519E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	nCCI	SPY-E
6.357E-03	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
6.055E-03	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
5.995E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
5.947E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
5.298E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SPY-E
5.273E-03	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
5.202E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	WCCI	SRY-EL
5.153E-03	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	nSPY
5.111E-03	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
5.077E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	WCCI	SPY-E
4.568E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	WCCI	SRY-EL
4.490E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	WCCI	SPY-E
4.210E-03	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
4.129E-03	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.095E-03	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
4.001E-03	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	nSPY

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
3.719E-03	TB-S-nR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
3.663E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-22	nCCI	SPRY-L
3.604E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-22	nCCI	nSPY
3.388E-03	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
3.351E-03	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
3.072E-03	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
2.918E-03	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n3	DCCI	nSPY
2.822E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	WCCI	SRY-EL
2.785E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	nCCI	SRY-EL
2.754E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	nCCI	SPY-E
2.754E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	WCCI	SPY-E
2.724E-03	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
2.709E-03	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
2.613E-03	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
2.533E-03	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
2.201E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-24	WCCI	SRY-EL
2.156E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	nCCI	SRY-EL
2.150E-03	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
2.148E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-24	WCCI	SPY-E
2.129E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	nCCI	SPY-E
2.068E-03	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SPRY-L
2.048E-03	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
2.030E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-44	nCCI	SRY-EL
1.994E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-44	nCCI	SPY-E
1.922E-03	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.912E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.881E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	WCCI	SRY-EL
1.865E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.839E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	nCCI	SRY-EL
1.838E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	nCCI	SPY-E
1.836E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	WCCI	SPY-E
1.800E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	SRY-EL
1.778E-03	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
1.768E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	SPY-E
1.705E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	WCCI	SRY-EL
1.693E-03	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L
1.691E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
1.686E-03	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-pvn	RBB-n4	WCCI	SRY-EL
1.677E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
1.660E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	WCCI	SPY-E
1.659E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SPY-E
1.652E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	WCCI	SRY-EL
1.624E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	WCCI	SPY-E
1.532E-03	TB-S-nR	CFD-LV	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.494E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SPY-E

60

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
1.478E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.467E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-24	WCCI	SRY-EL
1.432E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-24	WCCI	SPY-E
1.422E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	nCCI	SRY-EL
1.419E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	nCCI	SPY-E
1.418E-03	TB-S-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.409E-03	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
1.382E-03	TB-S-nR	CFD-IP	CLC-nWD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.310E-03	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
1.307E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.302E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	DCCI	SPY-E
1.300E-03	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
1.294E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SPY-E
1.279E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-24	WCCI	SRY-EL
1.248E-03	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
1.248E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-24	WCCI	SPY-E
1.223E-03	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n2	DCCI	nSPY
1.153E-03	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SPY-E
1.137E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	WCCI	SRY-EL
1.107E-03	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	WCCI	SPY-E
1.104E-03	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
1.093E-03	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	nSPY
1.076E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	nCCI	SPRY-L
1.074E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	nCCI	nSPY
1.044E-03	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
1.009E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-44	nCCI	SPRY-L
1.009E-03	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-44	nCCI	nSPY
9.703E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	WCCI	SPY-E
9.703E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	WCCI	SRY-EL
9.515E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L
8.540E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
8.524E-04	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n4	WCCI	SPRY-L
8.510E-04	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
8.471E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	nSPY
8.389E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	WCCI	SPRY-L
8.377E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	WCCI	nSPY
7.854E-04	TB-S-nR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
7.757E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-22	nCCI	nSPY
7.757E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-22	nCCI	SPRY-L
7.547E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SPY-E
7.520E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SPRY-L
7.487E-04	TB-S-LR	CFD-n	CLC-nnn	CLV-000	CD-nILM	VBD-F	SPB-nn	RBB-nn	nCCI-LR	SPRY-L
7.464E-04	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-vn	RBB-nn	nCCI	SRY-EL
7.287E-04	TB-S-LR	CFD-LV	CLC-nDW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
7.273E-04	TB-S-nR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY

61

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
7.214E-04	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	nSPY
7.126E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	nCCI	SPRY-L
7.111E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	nCCI	nSPY
7.087E-04	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
7.059E-04	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
7.055E-04	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
7.016E-04	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	nSPY
6.575E-04	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-pvn	RBB-n2	WCCI	SPRY-L
6.566E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SPRY-L
6.512E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	nSPY
6.490E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
6.397E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	DCCI	SPY-E
6.325E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	nCCI-LR	nSPY
6.217E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	nCCI	SRY-EL
6.217E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPC-ny	RBB-22	nCCI	SPY-E
6.015E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	nCCI	SRY-EL
6.015E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	WCCI	SRY-EL
5.907E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	nCCI	SPY-E
5.907E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	WCCI	SPY-E
5.802E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	nSPY
5.781E-04	TB-S-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILH	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
5.558E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	WCCI	SPRY-L
5.547E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	WCCI	nSPY
5.440E-04	TB-S-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
5.236E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	SPY-E
5.236E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	SRY-EL
5.036E-04	TB-S-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILH	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
4.923E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-44	nCCI	SPY-E
4.923E-04	TB-S-EA	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-44	nCCI	SRY-EL
4.883E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
4.849E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	WCCI	SPRY-L
4.849E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-22	WCCI	SRY-EL
4.848E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-22	WCCI	SPY E
4.848E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	WCCI	nSPY
4.830E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	WCCI-LR	nSPY
4.654E-04	TB-S-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.606E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-24	WCCI	SPY-E
4.606E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-24	WCCI	SRY-EL
4.339E-04	TB-S-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILH	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.170E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n4	WCCI	SRY-EL
4.147E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-yy	RBB-n2	nCCI	SRY-EL
4.084E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	WCCI	SPY-E
4.084E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	WCCI	SRY-EL
3.970E-04	TB-S-nR	CFD-n	CLC-nnW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
3.938E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	nCCI	SRY-EL

62

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interaction	Drywell Sprays
3.938E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	WCCI	SPY-E
3.938E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	WCCI	SRY-EL
3.938E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	nCCI	SPY-E
3.820E-04	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.805E-04	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.767E-04	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	nSPY
3.732E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	nCCI	SPY-E
3.880E-04	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	DCCI	nSPY
3.676E-04	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	DCCI	nSPY
3.674E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	DCCI	SPY-E
3.649E-04	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-pvn	RBB-nn	nCCI	SPRY-L
3.649E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	SPY-E
3.612E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	SRY-EL
3.600E-04	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n4	WCCI	SRY-EL
3.583E-04	TB-S-nR	CFD-IP	CLC-nWD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
3.509E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-44	WCCI	SRY-EL
3.509E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-44	WCCI	SPY-E
3.491E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	nCCI	SRY-EL
3.491E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	nCCI	S[illegible]-E
3.463E-04	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
3.438E-04	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	nSPY
3.433E-04	TB-S-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
3.427E-04	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-yy	RBB-n4	WCCI	SRY-EL
3.383E-04	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n4	DCCI	nSPY
3.260E-04	TB-S-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-yy	RBB-n2	WCCI	SRY-EL
3.260E-04	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-yy	RBB-n2	WCCI	SRY-EL
3.251E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	SPY-E
3.234E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	WCCI	SRY-EL
3.176E-04	TB-S-nR	CFD-LT	CLC-nWD	CLV-023	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
3.072E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-24	WCCI	SPY-E
3.072E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-24	WCCI	SRY-EL
3.065E-04	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-yy	RBB-n3	WCCI	SRY-EL
2.979E-04	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.968E-04	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.952E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	WCCI	SPRY-L
2.952E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	nCCI	SPRY-L
2.952E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	WCCI	nSPY
2.952E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	nCCI	nSPY
2.938E-04	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	nSPY
2.911E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	WCCI	SPY-E
2.882E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	WCCI	SRY-EL
2.875E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	DCCI	SPY-E
2.818E-04	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-pvn	RBB-n2	WCCI	SRY-EL
2.737E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-24	WCCI	SRY-EL
2.737E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-24	WCCI	SPY-E

63

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interaction	Drywell Sprays
2.723E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	WCCI	SPY-E
2.723E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	WCCI	SRY-EL
2.681E-04	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	nSPY
2.677E-04	TB-S-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
2.593E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	WCCI	SPY-E
2.564E-04	TB-S-nR	CFD-n	CLC-nWD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
2.426E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-22	WCCI	SPY-E
2.426E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-22	WCCI	SRY-EL
2.382E-04	TB-S-nR	CFD-n	CLC-nWW	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
2.327E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	nCCI	SPRY-L
2.327E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	nCCI	nSPY
2.319E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	nCCI-LR	SPRY-L
2.303E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-24	WCCI	nSPY
2.303E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-24	WCCI	SPRY-L
2.297E-04	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
2.296E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	nCCI-LR	nSPY
2.278E-04	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
2.254E-04	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n3	DCCI	nSPY
2.206E-04	TB-S-nR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
2.199E-04	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n3	DCCI	nSPY
2.188E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-44	nCCI	SPRY-L
2.188E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-44	nCCI	nSPY
2.175E-04	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
2.153E-04	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	nSPY
2.048E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SPRY-L
2.041E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SPRY-L
1.991E-04	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pv	RBB-n4	DCCI	nSPY
1.998E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	nCCI	nSPY
1.968E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	WCCI	SPRY-L
1.968E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	WCCI	nSPY
1.968E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	nCCI	SPRY-L
1.957E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-22	nCCI	nSPY
1.957E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-22	nCCI	SPRY-L
1.939E-04	TB-S-ZR	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	SPRY-L
1.939E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	nSPY
1.855E-04	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-pvn	RBB-n4	WCCI	SPRY-L
1.843E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	nSPY
1.837E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	nSPY
1.837E-04	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
1.626E-04	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.825E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.818E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SPRY-L
1.815E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	WCCI	nSPY
1.815E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	WCCI	SPRY-L
1.804E-04	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	DCCI	SPY-E

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
1.779E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-22	nCCI	SRY-EL
1.779E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-22	nCCI	SPY-E
1.754E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-44	WCCI	SPY-E
1.754E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	nCCI	SPY-E
1.754E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	nCCI	SRY-EL
1.754E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	WCCI	SPRY-L
1.754E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-44	WCCI	SRY-EL
1.754E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	WCCI	nSPY
1.723E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	WCCI-LR	SPRY-L
1.706E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	WCCI-LR	nSPY
1.673E-04	TB-S-nR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-pvn	RBB-n4	DCCI	nSPY
1.642E-04	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	nCCI	SPY-E
1.636E-04	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	nSPY
1.631E-04	TB-S-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n4	WCCI-LR	SPRY-L
1.610E-04	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
1.597E-04	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SPRY-L
1.593E-04	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	nSPY
1.551E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	nCCI	nSPY
1.551E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	nCCI	SPRY-L
1.535E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-24	WCCI	SPRY-L
1.535E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-24	WCCI	nSPY
1.532E-04	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
1.477E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	nCCI	SPY-E
1.477E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	WCCI	SPY-E
1.477E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	WCCI	SRY-EL
1.477E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	nCCI	SRY-EL
1.456E-04	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
1.456E-04	TB-S-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
1.438E-04	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	nSPY
1.423E-04	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	DCCI	SPY-E
1.423E-04	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SPRY-L
1.421E-04	TB-S-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
1.404E-04	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SPRY-L
1.382E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	nCCI	SPY-E
1.382E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	nCCI	SRY-EL
1.358E-04	TB-S-ER	CFD-n	CLC-nnW	CLV-002	CD-ILM	VBD-F	SPB-vn	RBB-n2	WCCI	SRY-EL
1.349E-04	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.337E-04	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.336E-04	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
1.336E-04	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	nSPY
1.309E-04	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
1.281E-04	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	nSPY
1.273E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-24	WCCI	SRY-EL
1.273E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-24	WCCI	SPY-E
1.273E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-24	WCCI	SPRY-L

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
1.273E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-24	WCCI	nSPY
1.264E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	WCCI-LR	SPPY-L
1.251E-04	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	WCCI-LR	nSPY
1.245E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SPY-E
1.245E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SRY-EL
1.210E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	WCCI	SPRY-L
1.210E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	WCCI	nSPY
1.152E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-24	WCCI	SRY-EL
1.152E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-24	WCCI	SPY-E
1.078E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	WCCI	SPRY-L
1.078E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-22	WCCI	SRY-EL
1.078E-04	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	WCCI	nSPY
1.078E-04	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-22	WCCI	SPY-E
1.076E-04	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-nn	RBB-n2	DCCI	nSPY
1.062E-04	TB-S-LR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
1.022E-04	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-pvy	RBB-n2	DCCI	nSPY
9.780E-05	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
9.719E-05	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvn	RBB-n4	DCCI	nSPY
9.657E-05	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
9.561E-05	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	nSPY
9.555E-05	TB-S-LR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	nSPY
9.196E-05	TB-S-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-yy	RBB-n4	WCCI	SRY-EL
9.123E-05	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n4	WCCI	SRY-EL
9.038E-05	TB-S-nR	CFD-n	CLC-nWW	CLV-022	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
8.861E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-44	WCCI	SRY-EL
8.861E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-44	WCCI	SPY-E
8.861E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-44	nCCI	SPY-E
8.861E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-44	nCCI	SRY-EL
8.736E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	WCCI	SPY-E
8.736E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	WCCI	SRY-EL
8.483E-05	TB-S-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SPRY-L
8.407E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
8.403E-05	TB-S-LR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
8.217E-05	TB-S-nR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
8.210E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n4	WCCI	SPY-E
8.151E-05	TB-S-ER	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	[illegible]	RBB-n2	WCCI	SRY-EL
8.127E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	[illegible]	RBB-n4	WCCI	SRY-EL
8.127E-05	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH-B	[illegible]	RBB-n2	DCCI	nSPY
8.110E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	[illegible]	RBB-n4	DCCI	SPY-E
7.947E-05	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	[illegible]	RBB-n4	WCCI	SRY-EL
7.894E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-44	WCCI	SPY-E
7.894E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-44	WCCI	SRY-EL
7.635E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	nSPY
7.563E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	nSPY
7.475E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	SPRY-L

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
7.469E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
7.467E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SRY-EL
7.467E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SPY-E
7.400E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	nSPY
7.315E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n4	WCCI	SPY-E
7.231E-05	TB-S-nR	CFD-LT	CLC-nWD	CLV-023	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
7.168E-05	TB-S-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n4	WCCI-LR	SPRY-L
7.007E-05	TB-S-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
7.007E-05	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
6.911E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-24	WCCI	SRY-EL
6.911E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-24	WCCI	SPY-E
6.756E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
6.722E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
6.688E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	nSPY
6.469E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	DCCI	SPY-E
6.158E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-24	WCCI	SPY-E
6.158E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-24	WCCI	SRY-EL
6.126E-05	TB-S-nR	CFD-IP	CLC-nWD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
6.064E-05	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
6.064E-05	TB-S-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
5.955E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
5.907E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	WCCI	nSPY
5.907E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	nCCI	nSPY
5.907E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	WCCI	SPRY-L
5.907E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	nCCI	SPRY-L
5.873E-05	TB-S-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
5.873E-05	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
5.831E-05	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
5.826E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
5.824E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	WCCI	SPY-E
5.824E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	WCCI	SRY-EL
5.784E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-22	nCCI	SRY-EL
5.784E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-22	nCCI	SPY-E
5.784E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	nCCI	SPRY-L
5.784E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	nCCI	nSPY
5.711E-05	TB-S-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
5.612E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
5.423E-05	TB-S-nR	CFD-n	CLC-nWW	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
5.284E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	nCCI	nSPY
5.284E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	nCCI	SPRY-L
5.243E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
5.236E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	SPRY-L
5.236E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	nSPY
5.212E-05	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	DCCI	nSPY
5.191E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
5.189E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-22	WCCI	SPY-E
5.189E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-22	WCCI	SRY-EL
5.183E-05	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
5.051E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SPY-E
5.019E-05	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
4.999E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
4.968E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-44	nCCI	nSPY
4.968E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-44	nCCI	SPRY-L
4.930E-05	TB-S-nR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.923E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-44	nCCI	SPRY-L
4.923E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-44	nCCI	nSPY
4.890E-05	TB-S-ER	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-yy	RBB-n2	WCCI	SRY-EL
4.803E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	nCCI	SRY-EL
4.803E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	nCCI	SPY-E
4.759E-05	TB-S-nR	CFD-n	CLC-nnW	CLV-002	CD-nILM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
4.692E-05	TB-S-nR	CFD-n	CLC-nnW	CLV-002	CD-nIHM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
4.671E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SPY-E
4.665E-05	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
4.608E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-24	WCCI	SPRY-L
4.608E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-24	WCCI	nSPY
4.566E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
4.519E-05	TB-S-nR	CFD-n	CLC-nWD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.517E-05	TB-S-ER	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
4.516E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-44	nCCI	SPY-E
4.516E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-44	nCCI	SRY-EL
4.512E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-22	WCCI	SPRY-L
4.512E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-22	WCCI	nSPY
4.505E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	SPRY-L
4.378E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
4.285E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	nSPY
4.121E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	WCCI	nSPY
4.121E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	WCCI	SPRY-L
4.105E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	nCCI	SRY-EL
4.084E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	WCCI	nSPY
4.084E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	WCCI	SPRY-L
4.055E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	nSPY
4.054E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-pvn	RBB-nn	nCCI	SRY-EL
4.014E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	SPRY-L
3.940E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SPY-E
3.938E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	WCCI	SPRY-L
3.938E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	WCCI	nSPY
3.938E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	nCCI	nSPY
3.938E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	nCCI	SPRY-L
3.771E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
3.747E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	WCCI	SPY-E

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
3.747E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	WCCI	SRY-EL
3.694E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	nCCI	SPY-E
3.675E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	WCCI-LR	SPRY-L
3.667E-05	TB-S-nR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-DH	SPB-nn	RBB-n4	DCCI	nSPY
3.664E-05	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n4	WCCI-LR	SPRY-L
3.658E-05	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n4	WCCI	SPRY-L
3.639E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	WCCI-LR	nSPY
3.628E-05	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n4	WCCI-LR	nSPY
3.612E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	nSPY
3.612E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	DCCI	SPY-E
3.594E-05	TB-S-nR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-pvn	RBB-n4	DCCI	nSPY
3.523E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	nCCI	SPRY-L
3.523E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	nCCI	nSPY
3.509E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-44	nCCI	SPY-E
3.509E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-44	WCCI	SRY-EL
3.509E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-44	WCCI	SPRY-L
3.509E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-44	WCCI	SPY-E
3.509E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-44	nCCI	SRY-EL
3.509E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-44	WCCI	nSPY
3.491E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	nCCI	SPRY-L
3.491E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	nCCI	nSPY
3.454E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
3.300E-05	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-DH	SPB-ny	RBB-n3	DCCI	nSPY
3.288E-05	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
3.253E-05	TB-S-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
3.213E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
3.202E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	nCCI	SPY-E
3.202E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	nCCI	SRY-EL
3.195E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
3.159E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	nCCI	SRY-EL
3.159E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	nCCI	SPY-E
3.159E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	WCCI	SRY-EL
3.159E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	WCCI	SPY-E
3.153E-05	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
3.110E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-22	nCCI	SRY-EL
3.110E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-22	nCCI	SPY-E
3.109E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SPY-E
3.100E-05	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-pvn	RBB-n2	WCCI	SPRY-L
3.072E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-24	WCCI	nSPY
3.072E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-24	WCCI	SPRY-L
2.919E-05	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	WCCI-LR	SPRY-L
2.919E-05	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	nCCI-LR	SPRY-L
2.889E-05	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	WCCI-LR	nSPY
2.889E-05	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	nCCI-LR	nSPY
2.882E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	DCCI	SPY-E

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
2.856E-05	TB-S-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
2.815E-05	TB-S-nR	CFD-LP	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
2.748E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	WCCI	SPRY-L
2.748E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	WCCI	nSPY
2.737E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-24	WCCI	nSPY
2.737E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-24	WCCI	SPRY-L
2.737E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-24	WCCI	SRY-EL
2.737E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-24	WCCI	SPY-E
2.723E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	WCCI	SPRY-L
2.723E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	WCCI	nSPY
2.694E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
2.676E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-pvy	RBB-22	nCCI	SPY-E
2.676E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-pvy	RBB-22	nCCI	SRY-EL
2.512E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
2.508E-05	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-pvy	RBB-n4	DCCI	nSPY
2.498E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	WCCI	SRY-EL
2.498E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	WCCI	SPY-E
2.489E-05	TB-S-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
2.464E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-24	WCCI	SRY-EL
2.464E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-24	WCCI	SPY-E
2.450E-05	TB-S-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-nn	RBB-n4	DCCI	nSPY
2.448E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-yy	RBB-22	WCCI	nSPY
2.448E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-yy	RBB-22	WCCI	SPRY-L
2.438E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
2.426E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-22	WCCI	SPRY-L
2.426E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-22	WCCI	nSPY
2.426E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-vn	RBB-22	WCCI	SRY-EL
2.426E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-vn	RBB-22	WCCI	SPY-E
2.425E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SPY-E
2.356E-05	TB-S-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
2.356E-05	TB-S-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
2.299E-05	TB-S-ER	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-vn	RBB-n4	WCCI	SRY-EL
2.292E-05	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH-B	SPB-ny	RBB-n4	DCCI	nSPY
2.261E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	nSPY
2.235E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	DCCI	SPY-E
2.225E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-yy	RBB-22	WCCI	SPY-E
2.225E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-yy	RBB-22	WCCI	SRY-EL
2.199E-05	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
2.177E-05	TB-S-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	nSPY
2.155E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
2.109E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n2	nCCI-LR	SPRY-L
2.106E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	WCCI	SPY-E
2.106E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	nCCI	SPY-E
2.106E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	nCCI	SRY-EL
2.106E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	WCCI	SRY-EL

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
2.068E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n2	nCCI-LR	nSPY
2.068E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-pvy	RBB-22	WCCI	SPY-E
2.068E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-pvy	RBB-22	WCCI	SRY-EL
2.027E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	nCCI	SPRY-L
2.009E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SPRY-L
1.939E-05	TB-S-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n2	WCCI	SPY-E
1.921E-05	TB-S-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-S	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
1.876E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	WCCI	SPY-E
1.876E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	WCCI	SRY-EL
1.866E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SPRY-L
1.842E-05	TB-S-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.842E-05	TB-S-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
1.832E-05	TB-S-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
1.825E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	nCCI	nSPY
1.814E-05	TB-S-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	nSPY
1.785E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.784E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-pvy	RBB-22	nCCI	SPY-E
1.784E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-pvy	RBB-22	nCCI	SRY-EL
1.743E-05	TB-S-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	nCCI	nSPY
1.702E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
1.685E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	nSPY
1.660E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
1.658E-05	TB-S-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SPY-E
1.658E-05	TB-S-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SPY-E
1.645E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n2	WCCI-LR	SPRY-L
1.643E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.643E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPY-E
1.643E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-24	WCCI	SRY-EL
1.631E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-44	WCCI	SPRY-L
1.631E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-44	nCCI	SPRY-L
1.631E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-44	nCCI	SPY-E
1.631E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-44	WCCI	nSPY
1.631E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-44	nCCI	SRY-EL
1.631E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-44	nCCI	nSPY
1.629E-05	TB-S-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n2	WCCI-LR	nSPY
1.612E-05	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n4	WCCI-LR	SPRY-L
1.606E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SPY-E
1.596E-05	TB-S-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n4	WCCI-LR	nSPY
1.571E-05	TB-S-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	nCCI-LR	SPRY-L
1.490E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	nCCI	SPRY-L
1.490E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	WCCI	nSPY
1.490E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	nCCI	nSPY
1.490E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	WCCI	SPRY-L
1.479E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
1.479E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L

71

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
1.477E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	WCCI	SPRY-L
1.477E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	nCCI	SPRY-L
1.477E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	nCCI	nSPY
1.477E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	WCCI	nSPY
1.464E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
1.464E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-24	WCCI	SRY-EL
1.464E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-24	WCCI	SPY-E
1.429E-05	TB-S-LR	CFD-LV	CLC-nWW	CLV-023	CD-nINM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
1.423E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	DCCI	nSPY
1.415E-05	TB-S-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	nSPY
1.392E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.392E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-pvy	RBB-22	WCCI	SPY-E
1.392E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-pvy	RBB-22	WCCI	SRY-EL
1.390E-05	TB-S-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	nSPY
1.382E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	nCCI	SPRY-L
1.382E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	nCCI	nSPY
1.382E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-22	nCCI	SRY-EL
1.382E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-22	nCCI	SPY-E
1.355E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	WCCI	SPY-E
1.355E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	nCCI	SRY-EL
1.355E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	nCCI	SPY-E
1.355E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	WCCI	SRY-EL
1.318E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n4	WCCI	SPY-E
1.313E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	nCCI	SRY-EL
1.313E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	nCCI	SPY-E
1.313E-05	TB-S-ER	CFD-IP	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	nCCI	SPY-E
1.313E-05	TB-S-ER	CFD-IP	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	nCCI	SRY-EL
1.273E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-24	WCCI	SPRY-L
1.273E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-24	WCCI	nSPY
1.269E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.253E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	WCCI	SPY-E
1.245E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
1.245E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.240E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.240E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-pvy	RBB-22	WCCI	SPY-E
1.240E-05	TB-S-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-pvy	RBB-22	WCCI	SRY-EL
1.225E-05	TB-S-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	WCCI-LR	SPRY-L
1.206E-05	TB-S-nR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-S	SPB-nn	RBB-n4	DCCI	nSPY
1.162E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-24	WCCI	SPRY-L
1.162E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-24	WCCI	nSPY
1.152E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-24	WCCI	nSPY
1.152E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-24	WCCI	SPRY-L
1.142E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	nCCI-LR	nSPY
1.116E-05	TB-S-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	WCCI	SPY-E
1.099E-05	TB-S-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L

Table A-2
Central Estimate Detailed Results For The TB Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Drywell Sprays
1.088E-05	TB-S-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	nSPY
1.078E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-pvn	RBB-22	WCCI	SPRY-L
1.078E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-pvn	RBB-22	WCCI	nSPY
1.057E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-24	WCCI	SPY-E
1.057E-05	TB-S-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-24	WCCI	SRY-EL
1.025E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	nCCI-LR	SPRY-L
1.024E-05	TB-S-ER	CFD-E4V	CLC-WDU	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	WCCI	SPY-E
1.024E-05	TB-S-ER	CFD-E4V	CLC-WDU	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	WCCI	SRY-EL
1.024E-05	TB-S-ER	CFD-IP	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	WCCI	SRY-EL
1.024E-05	TB-S-ER	CFD-IP	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	WCCI	SPY-E
1.024E-05	TB-S-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	nSPY
1.018E-05	TB-S-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	nCCI-LR	nSPY

9.8794E-01 TOTAL FREQUENCY OF BINS PRINTED

TOTAL FREQUENCY IN ALL BINS = 1.0000E+00
TOTAL FREQUENCY LOST DUE TO TERMINATION OF BRANCHES 1.2062E-02
TOTAL FREQUENCY FOR THIS SEQUENCE/WALKTHROUGH = 1.0121E+00

Table A-3
Central Estimate Detailed Results For The TBU Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Node	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.824E-01	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
7.412E-02	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
6.303E-02	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
6.302E-02	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
5.561E-02	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
5.144E-02	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.103E-02	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
3.611E-02	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
3.592E-02	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.556E-02	TB-F-LR	CFD-n	CLC-nnn	CLV-000	CD-nILM	VBD-F	SPB-nn	RBB-nn	nCCI-LR	SPRY-L
3.556E-02	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
2.802E-02	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.616E-02	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
2.616E-02	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
2.104E-02	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
1.458E-02	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.374E-02	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.374E-02	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.274E-02	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
1.132E-02	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
1.132E-02	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
1.018E-02	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
9.980E-03	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
8.735E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
8.734E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
7.899E-03	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
7.542E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
7.541E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
6.959E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
6.959E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
6.363E-03	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n3	DCCI	nSPY
4.111E-03	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
3.871E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
3.871E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
2.888E-03	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
2.888E-03	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
2.815E-03	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
2.464E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
2.464E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
1.783E-03	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.641E-03	TB-F-nR	CFD-n	CLC-nnW	CLV-002	CD-nILM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
1.595E-03	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
1.510E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SPRY-L
1.510E-03	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SPRY-L
1.415E-03	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	nCCI-LR	SPRY-L

Table A-3
Central Estimate Detailed Results For The TBU Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.401E-03	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-S	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
1.235E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L
1.235E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L
1.235E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
1.235E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
1.223E-03	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
1.223E-03	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
1.203E-03	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
9.848E-04	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
9.538E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SPRY-L
9.538E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SPRY-L
8.981E-04	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
8.235E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L
8.235E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L
6.532E-04	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
5.037E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
5.030E-04	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.606E-04	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-S	SPB-nn	RBB-n4	DCCI	nSPY
4.396E-04	TB-F-nR	CFD-n	CLC-nnW	CLV-002	CD-nILM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
4.228E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SPRY-L
4.228E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SPRY-L
3.539E-04	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
3.470E-04	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
3.461E-04	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.190E-04	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
2.874E-04	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
2.757E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	DCCI	nSPY
2.753E-04	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
2.700E-04	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.690E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n4	WCCI	SPRY-L
2.690E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SPRY-L
2.638E-04	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
1.936E-04	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.770E-04	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.730E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nILM	VBD-SE	SPB-ny	RBB-n2	DCCI	nSPY
1.730E-04	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-SE	SPB-ny	RBB-n2	DCCI	nSPY
1.697E-04	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	nSPY
1.697E-04	TB-F-LR	CFD-n	CLC-nnn	CLV-000	CD-nILM	VBD-F	SPB-nn	RBB-nn	nCCI-LR	nSPY
1.680E-04	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	nSPY
1.583E-04	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
1.512E-04	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
1.512E-04	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
1.479E-04	TB-F-LR	CFD-n	CLC-nnn	CLV-000	CD-nILM	VBD-S	SPB-nn	RBB-nn	nCCI-LR	SPRY-L
1.329E-04	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-S	SPB-ny	RBB-n3	DCCI	nSPY
1.324E-04	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	nSPY

Table A-3
Central Estimate Detailed Results For The TBU Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.303E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
1.303E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
1.290E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SPRY-L
1.290E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	nCCI	SPRY-L
1.277E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-S	SPB-nn	RBB-n2	WCCI	SPRY-L
1.277E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SPRY-L
1.097E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.090E-04	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
1.087E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.081E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
9.788E-05	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
8.599E-05	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
7.808E-05	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
7.808E-05	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
7.728E-05	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
7.438E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
7.327E-05	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
6.735E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F-B	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
6.735E-05	TB-F-LR	CFD-n	CLC-nnn	CLV-000	CD-nILM	VBD-F-B	SPB-ny	RBB-nn	nCCI-LR	SPRY-L
6.291E-05	TB-F-LR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
6.291E-05	TB-F-LR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
6.184E-05	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S-B	SPB-ny	RBB-n2	DCCI	nSPY
6.146E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
5.802E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
5.672E-05	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
5.377E-05	TB-F-LR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n3	WCCI-LR	SPRY-L
5.324E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
5.253E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F-B	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
5.201E-05	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F-B	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
5.075E-05	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
4.879E-05	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-SE	SPB-ny	RBB-n4	DCCI	nSPY
4.879E-05	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nILM	VBD-SE	SPB-ny	RBB-n4	DCCI	nSPY
4.840E-05	TB-F-LR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
4.779E-05	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	DCCI	nSPY
4.718E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	nSPY
4.658E-05	TB-F-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
4.658E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
4.658E-05	TB-F-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
4.658E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
4.560E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
4.519E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
4.153E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
4.030E-05	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.004E-05	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
3.981E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL

Table A-3
Central Estimate Detailed Results For The TBU Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
3.981E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
3.755E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
3.755E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
3.734E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	nSPY
3.583E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
3.583E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
3.357E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nILM	VBD-SE	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.357E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-SE	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.291E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	nSPY
3.291E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	nSPY
3.268E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.036E-05	TB-F-nR	CFD-n	CLC-nnW	CLV-002	CD-nILM	VBD-S	SPB-nn	RBB-n2	DCCI	nSPY
2.959E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
2.774E-05	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
2.742E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
2.621E-05	TB-F-LR	CFD-n	CLC-nnW	CLV-002	CD-nILM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
2.619E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nILM	VBD-SE	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.619E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-SE	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.618E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
2.565E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.395E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
2.374E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
2.306E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
2.216E-05	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n3	DCCI	nSPY
2.202E-05	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
2.042E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
2.017E-05	TB-F-LR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
1.991E-05	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
1.974E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.956E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.872E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S-B	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
1.851E-05	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-S	SPB-nn	RBB-n2	DCCI	nSPY
1.775E-05	TB-F-LR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
1.744E-05	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S-B	SPB-ny	RBB-n4	DCCI	nSPY
1.580E-05	TB-F-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.580E-05	TB-F-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.524E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.510E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.493E-05	TB-F-ER	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
1.493E-05	TB-F-ER	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
1.482E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F-B	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
1.467E-05	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F-B	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
1.434E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SPRY-L
1.434E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SPRY-L
1.369E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	nSPY

77

Table A-3
Central Estimate Detailed Results For The TBU Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.369E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	nSPY
1.331E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	nSPY
1.320E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.320E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.316E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.314E-05	TB-F-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.314E-05	TB-F-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.304E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.206E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SPRY-L
1.195E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-pvn	RBB-nn	nCCI	SPRY-L
1.140E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-vn	RBB-n2	WCCI	SRY-EL
1.140E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
1.140E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
1.111E-05	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
1.050E-05	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
1.050E-05	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L

9.9701E-01 TOTAL FREQUENCY OF BINS PRINTED

TOTAL FREQUENCY IN ALL BINS = 1.0000E+00
TOTAL FREQUENCY LOST DUE TO TERMINATION OF BRANCHES 2.9942E-03
TOTAL FREQUENCY FOR THIS SEQUENCE/WALKTHROUGH = 1.0030E+00

Table A-4
Central Estimate Detailed Results For The TBUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
2.441E-01	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
7.895E-02	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
6.884E-02	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.665E-02	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.420E-02	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	DCCI	nSPY
3.502E-02	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
2.972E-02	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
2.972E-02	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
2.584E-02	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
2.224E-02	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
2.002E-02	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.561E-02	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
1.325E-02	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
1.259E-02	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.256E-02	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	DCCI	nSPY
1.230E-02	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
1.219E-02	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
1.177E-02	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
1.061E-02	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
9.315E-03	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
9.315E-03	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
9.180E-03	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
8.017E-03	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
6.641E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
6.641E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
6.454E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
6.399E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
6.374E-03	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
5.319E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
5.274E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
5.116E-03	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
4.972E-03	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
4.376E-03	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
4.107E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
4.072E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
4.008E-03	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n3	DCCI	nSPY
3.838E-03	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
3.781E-03	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
3.751E-03	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
3.558E-03	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.551E-03	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
3.546E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
3.516E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
3.320E-03	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
3.281E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L

Table A-4
Central Estimate Detailed Results For The TBUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
3.281E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
3.001E-03	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
2.832E-03	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILH	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
2.757E-03	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.748E-03	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
2.723E-03	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
2.569E-03	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
2.402E-03	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nILH	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
2.219E-03	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
2.087E-03	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n2	DCCI	nSPY
1.988E-03	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.820E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.805E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.803E-03	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
1.661E-03	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
1.652E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
1.453E-03	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nILH	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
1.443E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.430E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.430E-03	TB-F-nR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
1.376E-03	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
1.352E-03	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
1.340E-03	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
1.318E-03	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.280E-03	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	SRY-EL
1.246E-03	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
1.190E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.179E-03	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.158E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
1.149E-03	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
1.126E-03	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n3	DCCI	nSPY
1.058E-03	TB-F-nR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
1.055E-03	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
1.028E-03	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nILH	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.021E-03	TB-F-nR	CFD-n	CLC-nnW	CLV-002	CD-nIHM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
1.018E-03	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILH	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
1.018E-03	TB-F-LR	CFD-n	CLC-nnn	CLV-000	CD-nILH	VBD-F	SPB-nn	RBB-nn	nCCI-LR	SPRY-L
1.006E-03	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILH	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
9.974E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
9.514E-04	TB-F-LR	CFD-IN	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	nCCI-LR	SPRY-L
9.184E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
9.101E-04	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
8.924E-04	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
8.924E-04	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
8.796E-04	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILH	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY

80

Table A-4
Central Estimate Detailed Results For The TBUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
8.363E-04	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
7.937E-04	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
7.930E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
7.858E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
7.582E-04	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
7.421E-04	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	WCCI-LR	SPRY-L
7.306E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SPRY-L
7.289E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SPRY-L
7.099E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SPRY-L
7.039E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SPRY-L
7.029E-04	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
6.869E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-vn	RBB-n2	WCCI	SRY-EL
6.391E-04	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-pvn	RBB-n4	DCCI	nSPY
6.282E-04	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	DCCI	nSPY
6.128E-04	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
5.851E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L
5.850E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
5.801E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L
5.801E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
5.792E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
5.777E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n4	DCCI	nSPY
5.743E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
5.445E-04	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
5.002E-04	TB-F-nR	CFD-IP	CLC-nWD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
4.947E-04	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
4.518E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SPRY-L
4.479E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SPRY-L
4.432E-04	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	WCCI-LR	SPRY-L
4.432E-04	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	nCCI-LR	SPRY-L
4.426E-04	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
4.424E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
4.392E-04	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-F	SPB-ny	RBB-n3	DCCI	nSPY
4.257E-04	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
4.138E-04	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	WCCI	SRY-EL
4.071E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	SRY-EL
4.034E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	SRY-EL
3.901E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L
3.868E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L
3.848E-04	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
3.802E-04	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
3.608E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	WCCI	SRY-EL
3.564E-04	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-44	nCCI	SRY-EL
3.022E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-pvn	RBB-n2	WCCI	SPRY-L
2.995E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-pvn	RBB-n2	WCCI	SPRY-L
2.974E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-vn	RBB-nn	nCCI	SRY-EL

81

Table A-4
Central Estimate Detailed Results For The TBUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
2.943E-04	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	nCCI-LR	SPRY-L
2.838E-04	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
2.829E-04	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
2.760E-04	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	WCCI	SRY-EL
2.734E-04	TB-F-nR	CFD-n	CLC-nnW	CLV-002	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
2.663E-04	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
2.663E-04	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
2.628E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
2.590E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	SRY-EL
2.567E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n4	WCCI	SRY-EL
2.29[illegible]E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	WCCI	SRY-EL
2.282E-04	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
2.274E-04	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	SRY-EL
2.239E-04	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
2.215E-04	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	WCCI-LR	SPRY-L
2.093E-04	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	WCCI-LR	SPRY-L
2.002E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SPRY-L
1.985E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SPRY-L
1.982E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	nCCI	SRY-EL
1.928E-04	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH-B	SPB-ny	RBB-n2	DCCI	nSPY
1.837E-04	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-nn	RBB-n2	DCCI	nSPY
1.817E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-vn	RBB-nn	nCCI	SPRY-L
1.812E-04	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-pvy	RBB-n3	DCCI	nSPY
1.809E-04	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-pvy	RBB-n2	DCCI	nSPY
1.656E-04	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n2	nCCI-LR	SPRY-L
1.640E-04	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
1.588E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SPRY-L
1.581E-04	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-22	WCCI	SRY-EL
1.573E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	WCCI	SPRY-L
1.516E-04	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.365E-04	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	nCCI	SRY-EL
1.357E-04	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	DCCI	nSPY
1.347E-04	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-44	WCCI	SRY-EL
1.333E-04	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.308E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-pvn	RBB-nn	nCCI	SPRY-L
1.308E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.297E-04	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-pvn	RBB-nn	nCCI	SPRY-L
1.296E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.295E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-pvn	RBB-n2	WCCI	SRY-EL
1.284E-04	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-S	SPB-pvn	RBB-n2	WCCI	SRY-EL
1.262E-04	TB-F-nR	CFD-n	CLC-nWW	CLV-022	CD-nIHM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
1.251E-04	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	SPRY-L
1.213E-04	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n2	WCCI-LR	SPRY-L
1.201E-04	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
1.192E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SPRY-L

Table A-4
Central Estimate Detailed Results For The TBUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.183E-04	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
1.182E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SPRY-L
1.170E-04	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
1.139E-04	TB-F-nR	CFD-IP	CLC-nWD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
1.133E-04	TB-F-nR	CFD-n	CLC-nnW	CLV-002	CD-nILM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
1.114E-04	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F-B	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.102E-04	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	DCCI	nSPY
1.051E-04	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-24	WCCI	SRY-EL
1.026E-04	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
1.018E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n4	WCCI	SRY-EL
1.010E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SPRY-L
1.009E-04	TB-F-nR	CFD-LT	CLC-nWD	CLV-023	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
1.001E-04	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SPRY-L
9.117E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	nSPY
8.981E-05	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-44	WCCI	SRY-EL
8.723E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SPRY-L
8.705E-05	TB-F-LR	CFD-n	CLC-nnn	CLV-000	CD-nIHM	VBD-F	SPB-nn	RBB-nn	nCCI-LR	SPRY-L
8.644E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SPRY-L
8.618E-05	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
8.140E-05	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
7.978E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
7.943E-05	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n4	DCCI	nSPY
7.846E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F-B	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
7.826E-05	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
7.570E-05	TB-F-nR	CFD-n	CLC-nWW	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
7.556E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-vn	RBB-n2	WCCI	SPRY-L
7.005E-05	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-24	WCCI	SRY-EL
6.798E-05	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n2	DCCI	nSPY
6.761E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
6.591E-05	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	DCCI	nSPY
6.476E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n4	WCCI	SRY-EL
6.308E-05	TB-F-nR	CFD-n	CLC-nWD	CLV-022	CD-nIHM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
6.274E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	nSPY
6.259E-05	TB-F-nR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-DH	SPB-nn	RBB-n4	DCCI	nSPY
6.083E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
6.064E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
6.029E-05	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-pvy	RBB-n2	DCCI	nSPY
6.028E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
6.022E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SPRY-L
6.003E-05	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F-B	SPB-pvy	RBB-n2	DCCI	nSPY
5.968E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SPRY-L
5.962E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SPRY-L
5.906E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SPRY-L
5.773E-05	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-44	nCCI	SRY-EL
5.726E-05	TB-F-nR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-pvn	RBB-n4	DCCI	nSPY

Table A-4
Central Estimate Detailed Results For The TBUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
5.715E-05	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-44	WCCI	SRY-EL
5.633E-05	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-DH	SPB-ny	RBB-n3	DCCI	nSPY
5.606E-05	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
5.606E-05	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
5.579E-05	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
5.573E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	WCCI-LR	SPRY-L
5.177E-05	TB-F-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
5.153E-05	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F	SPB-pvy	RBB-n3	DCCI	nSPY
4.673E-05	TB-F-nR	CFD-n	CLC-nnW	CLV-002	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	DCCI	nSPY
4.591E-05	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-pvy	RBB-n4	DCCI	nSPY
4.478E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n4	WCCI	SPRY-L
4.456E-05	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-24	WCCI	SRY-EL
4.437E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n4	WCCI	SPRY-L
4.395E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
4.326E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
4.183E-05	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-nn	RBB-n4	DCCI	nSPY
4.098E-05	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	WCCI	SPRY-L
4.079E-05	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-ny	RBB-n2	DCCI	nSPY
4.029E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-S	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
3.969E-05	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-S	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
3.969E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	WCCI	SPRY-L
3.916E-05	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH-B	SPB-ny	RBB-n4	DCCI	nSPY
3.907E-05	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-DH	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
3.907E-05	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-DH	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
3.849E-05	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-44	nCCI	SRY-EL
3.828E-05	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvn	RBB-n4	DCCI	nSPY
3.574E-05	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	WCCI-LR	SPRY-L
3.574E-05	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	nCCI-LR	SPRY-L
3.526E-05	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-44	nCCI	SPRY-L
3.513E-05	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	nSPY
3.513E-05	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	nSPY
3.500E-05	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-pvy	RBB-n2	DCCI	nSPY
3.471E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n4	WCCI	SRY-EL
3.271E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-vn	RBB-nn	nCCI	SPRY-L
3.271E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	nCCI	SRY-EL
3.238E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-vn	RBB-n2	WCCI	SRY-EL
3.235E-05	TB-F-nR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n4	DCCI	nSPY
3.226E-05	TB-F-nR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
3.226E-05	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nIHM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
3.029E-05	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-DH	SPB-pvy	RBB-n4	DCCI	nSPY
2.976E-05	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n2	nCCI	nSPY
2.940E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-pvy	RBB-n2	WCCI	SRY-EL
2.876E-05	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-S	SPB-nn	RBB-n4	DCCI	nSPY
2.849E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	SPRY-L
2.824E-05	TB-F-ER	CFD-IN	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n4	WCCI	SPRY-L

Table A-4
Central Estimate Detailed Results For The TBUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
2.804E-05	TB-F-nR	CFD-LP	CLC-nnW	CLV-003	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	DCCI	nSPY
2.746E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
2.732E-05	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	WCCI	SPRY-L
2.692E-05	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
2.526E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	WCCI	SPRY-L
2.523E-05	TB-F-nR	CFD-IM	CLC-nDD	CLV-033	CD-nILM	VBD-F-B	SPB-ny	RBB-n2	DCCI	nSPY
2.510E-05	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-ny	RBB-n4	DCCI	nSPY
2.464E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH	SPB-pvy	RBB-n4	WCCI-LR	SPRY-L
2.463E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
2.415E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
2.296E-05	TB-F-nR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvy	RBB-n4	DCCI	nSPY
2.295E-05	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-S	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
2.274E-05	TB-F-nR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F-B	SPB-ny	RBB-n4	DCCI	nSPY
2.252E-05	TB-F-ER	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	SPRY-L
2.251E-05	TB-F-ER	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-yy	RBB-22	nCCI	SRY-EL
2.244E-05	TB-F-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
2.244E-05	TB-F-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
2.241E-05	TB-F-ER	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F-B	SPB-ny	RBB-n3	nCCI	SRY-EL
2.181E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	nCCI	SPRY-L
2.174E-05	TB-F-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
2.174E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
2.155E-05	TB-F-ER	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
2.155E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
2.084E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n4	WCCI-LR	SPRY-L
2.053E-05	TB-F-LR	CFD-IM	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-pvy	RBB-n2	WCCI-LR	SPRY-L
2.046E-05	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-F-B	SPB-ny	RBB-n3	DCCI	nSPY
1.992E-05	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
1.858E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
1.849E-05	TB-F-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.842E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
1.673E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.657E-05	TB-F-ER	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.655E-05	TB-F-nR	CFD-LV	CLC-nnW	CLV-003	CD-nIHM	VBD-S	SPB-nn	RBB-n4	DCCI	nSPY
1.637E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
1.582E-05	TB-F-nR	CFD-LT	CLC-nnD	CLV-003	CD-nILM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
1.546E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.536E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	nSPY
1.522E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-nn	RBB-nn	nCCI	nSPY
1.454E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.441E-05	TB-F-ER	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-S	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.439E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	nCCI	SPRY-L
1.426E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	nCCI	SPRY-L
1.425E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-pvn	RBB-n2	WCCI	SPRY-L
1.419E-05	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F-B	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
1.419E-05	TB-F-LR	CFD-LV	CLC-nDW	CLV-023	CD-nIHM	VBD-F-B	SPB-ny	RBB-n2	nCCI-LR	SPRY-L

85

Table A-4
Central Estimate Detailed Results For The TBUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.412E-05	TB-F-ER	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-S	SPB-pvn	RBB-n2	WCCI	SPRY-L
1.390E-05	TB-F-nR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S-B	SPB-ny	RBB-n2	DCCI	nSPY
1.381E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
1.377E-05	TB-F-ER	CFD-IP	CLC-nWW	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.346E-05	TB-F-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.332E-05	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
1.332E-05	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-DH	SPB-nn	RBB-n2	WCCI-LR	SPRY-L
1.315E-05	TB-F-LR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
1.313E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.281E-05	TB-F-nR	CFD-n	CLC-nDD	CLV-022	CD-nIHM	VBD-S	SPB-ny	RBB-n4	DCCI	nSPY
1.237E-05	TB-F-ER	CFD-LV	CLC-nDW	CLV-023	CD-ILH	VBD-F-B	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.222E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
1.220E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S-B	SPB-ny	RBB-n4	WCCI	SRY-EL
1.219E-05	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	WCCI-LR	SPRY-L
1.219E-05	TB-F-LR	CFD-IP	CLC-nWW	CLV-033	CD-nIHM	VBD-F	SPB-pvn	RBB-n2	nCCI-LR	SPRY-L
1.146E-05	TB-F-nR	CFD-LP	CLC-nnD	CLV-003	CD-nIHM	VBD-S	SPB-ny	RBB-n2	DCCI	nSPY
1.124E-05	TB-F-LR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n3	WCCI-LR	SPRY-L
1.122E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-DH-B	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.119E-05	TB-F-ER	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n4	WCCI	SPRY-L
1.077E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
1.068E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
1.053E-05	TB-F-LR	CFD-IP	CLC-nDD	CLV-033	CD-nIHM	VBD-S	SPB-ny	RBB-n2	nCCI-LR	SPRY-L
1.050E-05	TB-F-LR	CFD-LV	CLC-nWW	CLV-023	CD-nIHM	VBD-F	SPB-ny	RBB-n2	WCCI-LR	SPRY-L
1.034E-05	TB-F-ER	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S-B	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.018E-05	TB-F-LR	CFD-LV	CLC-nnW	CLV-003	CD-nILM	VBD-F	SPB-nn	RBB-n2	nCCI-LR	SPRY-L
1.012E-05	TB-F-LR	CFD-n	CLC-nnD	CLV-002	CD-nIHM	VBD-F	SPB-ny	RBB-n4	WCCI-LR	SPRY-L

9.9402E-01 TOTAL FREQUENCY OF BINS PRINTED

TOTAL FREQUENCY IN ALL BINS = 1.0000E+00
TOTAL FREQUENCY LOST DUE TO TERMINATION OF BRANCHES 5.9797E-03
TOTAL FREQUENCY FOR THIS SEQUENCE/WALKTHROUGH = 1.0060E+00

86

Table A-5
Central Estimate Detailed Results For The TCUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.744E-01	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
1.361E-01	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.113E-01	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
1.113E-01	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
4.603E-02	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
4.562E-02	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
3.833E-02	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
3.069E-02	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-F	SPB-nn	RBB-n2	nCCI	SRY-EL
3.069E-02	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
2.416E-02	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
2.396E-02	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
2.161E-02	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	nCCI	SRY-EL
1.993E-02	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
1.975E-02	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
1.681E-02	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	WCCI	SRY-EL
1.539E-02	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
1.525E-02	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
1.329E-02	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
1.317E-02	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
9.142E-03	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
9.142E-03	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-nn	RBB-n2	nCCI	SRY-EL
6.813E-03	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
6.751E-03	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
5.785E-03	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
4.718E-03	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n4	WCCI	SRY-EL
4.513E-03	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
4.335E-03	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
4.296E-03	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
3.320E-03	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
2.190E-03	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
2.170E-03	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
2.168E-03	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
2.148E-03	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
1.260E-03	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
9.896E-04	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-F	SPB-nn	RBB-n2	nCCI	SRY-EL
9.896E-04	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
9.204E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
8.901E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
8.857E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
8.488E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
6.421E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
6.250E-04	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
5.524E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
4.893E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
4.875E-04	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL

87

Table A-5
Central Estimate Detailed Results For The TCUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
4.648E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
3.816E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
3.683E-04	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
3.649E-04	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
3.338E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
3.317E-04	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-DH	SPB-nn	RBB-n2	WCCI	SRY-EL
3.317E-04	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-DH	SPB-nn	RBB-n2	nCCI	SRY-EL
3.035E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
2.609E-04	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
2.666E-04	TC-F	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n3	nCCI	SRY-EL
2.411E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
2.408E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
2.389E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
2.292E-04	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
2.229E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
2.214E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
2.105E-04	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
1.934E-04	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.917E-04	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.886E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.669E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n4	WCCI	SRY-EL
1.594E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.580E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.490E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n4	WCCI	SRY-EL
1.477E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L
1.445E-04	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
1.403E-04	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.369E-04	TC-F	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
1.238E-04	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.231E-04	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.220E-04	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.185E-04	TC-F	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	SRY-EL
1.152E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SPRY-L
1.136E-04	TC-F	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.096E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
1.086E-04	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
1.076E-04	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
1.063E-04	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.053E-04	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
9.833E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-DH	SPB-nn	RBB-n2	WCCI	SRY-EL
9.833E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-DH	SPB-nn	RBB-n2	nCCI	SRY-EL
9.744E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
9.570E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
9.520E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH-B	SPB-ny	RBB-n2	nCCI	SRY-EL
9.206E-05	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-vn	RBB-n2	WCCI	SRY-EL

Table A-5
Central Estimate Detailed Results For The TCUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
9.059E-05	TC-F	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
8.977E-05	TC-F	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
8.883E-05	TC-F	CFD-n	CLC-nDD	CLV-022	CD-ILM	VBD-F-B	SPB-ny	RBB-n3	nCCI	SRY-EL
8.730E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
8.617E-05	TC-F	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
8.617E-05	TC-F	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
8.539E-05	TC-F	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
8.539E-05	TC-F	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
8.480E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	DCCI	SPY-E
7.905E-05	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
7.600E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
7.365E-05	TC-F	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
7.328E-05	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
7.299E-05	TC-F	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
6.937E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n3	WCCI	SRY-EL
6.937E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n3	nCCI	SRY-EL
6.852E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
6.457E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH-B	SPB-ny	RBB-n2	WCCI	SRY-EL
6.347E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-pvy	RBB-n4	WCCI	SRY-EL
6.227E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	nCCI	SPY-E
6.124E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
5.900E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	nCCI	SRY-EL
5.625E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
5.474E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
5.436E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	SRY-EL
5.406E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	SRY-EL
5.345E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
5.277E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
5.050E-05	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
5.050E-05	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-F	SPB-pvn	RBB-n2	nCCI	SRY-EL
5.047E-05	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
4.913E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n4	WCCI	SRY-EL
4.857E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	WCCI	SPY-E
4.836E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n4	WCCI	SRY-EL
4.836E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	WCCI	SRY-EL
4.797E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
4.602E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	WCCI	SRY-EL
4.557E-05	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
4.516E-05	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
4.270E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
4.116E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
3.989E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	DCCI	SPY-E
3.986E-05	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-vn	RBB-nn	nCCI	SRY-EL
3.883E-05	TC-F	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	WCCI	SRY-EL
3.788E-05	TC-F	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL

89

Table A-5
Central Estimate Detailed Results For The TCUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
3.591E-05	TC-F	CFD-n	CLC-nnW	CLV-002	CD-ILN	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
3.558E-05	TC-F	CFD-n	CLC-nnW	CLV-002	CD-ILH	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
3.472E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	SRY-EL
3.440E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n4	WCCI	SRY-EL
3.344E-05	TC-F	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-44	nCCI	SRY-EL
3.172E-05	TC-F	CFD-n	CLC-nDD	CLV-022	CD-ILM	VBD-F-B	SPB-ny	RBB-n3	nCCI	SRY-EL
3.118E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-ILN	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
3.077E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-yy	RBB-n2	WCCI	SRY-EL
2.955E-05	TC-F	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
2.812E-05	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
2.796E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-pvy	RBB-n2	nCCI	SRY-EL
2.762E-05	TC-F	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
2.737E-05	TC-F	CFD-LP	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
2.657E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n2	nCCI	SRY-EL
2.623E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SPRY-L
2.589E-05	TC-F	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	WCCI	SRY-EL
2.443E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S-B	SPB-ny	RBB-n2	nCCI	SRY-EL
2.431E-05	TC-F	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
2.409E-05	TC-F	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
2.394E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SPRY-L
2.372E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SPRY-L
2.344E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
2.194E-05	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
2.181E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHN	VBD-DH	SPB-pvy	RBB-n2	WCCI	SRY-EL
2.134E-05	TC-F	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	SRY-EL
2.083E-05	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
2.083E-05	TC-F	CFD-LV	CLC-nDW	CLV-023	CD-IHM	VBD-F	SPB-nn	RBB-n2	nCCI	SPRY-L
1.973E-05	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L
1.955E-05	TC-F	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L
1.821E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH-L	SPB-ny	RBB-n4	WCCI	SRY-EL
1.753E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.738E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.736E-05	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-pvn	RBB-n2	WCCI	SRY-EL
1.727E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n4	WCCI	SRY-EL
1.722E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
1.722E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-pvn	RBB-n2	nCCI	SRY-EL
1.720E-05	TC-F	CFD-LV	CLC-nnW	CLV-003	CD-ILH	VBD-S	SPB-pvn	RBB-n2	WCCI	SRY-EL
1.539E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n4	WCCI	SRY-EL
1.523E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SPRY-L
1.510E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SPRY-L
1.384E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	nCCI	SPRY-L
1.370E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n4	WCCI	SPY-E
1.364E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-yy	RBB-n4	WCCI	SRY-EL
1.315E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILN	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L
1.303E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L

Table A-5
Central Estimate Detailed Results For The TCUX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.296E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n4	WCCI	SRY-EL
1.293E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.293E-05	TC-F	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.281E-05	TC-F	CFD-LP	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.204E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n4	WCCI	SRY-EL
1.193E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-SE	SPB-ny	RBB-n4	WCCI	SRY-EL
1.161E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n4	WCCI	SRY-EL
1.158E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n2	WCCI	SPY-E
1.156E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
1.097E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
1.082E-05	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-DH	SPB-nn	RBB-n2	WCCI	SRY-EL
1.082E-05	TC-F	CFD-LV	CLC-nWW	CLV-023	CD-IHM	VBD-DH	SPB-nn	RBB-n2	nCCI	SRY-EL
1.079E-05	TC-F	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	WCCI	SPRY-L
1.071E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
1.042E-05	TC-F	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-yy	RBB-n2	WCCI	SRY-EL
1.033E-05	TC-F	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL

9.9779E-01 TOTAL FREQUENCY OF BINS PRINTED

TOTAL FREQUENCY IN ALL BINS = 1.0000E+00
TOTAL FREQUENCY LOST DUE TO TERMINATION OF BRANCHES 2.2125E-03
TOTAL FREQUENCY FOR THIS SEQUENCE/WALKTHROUGH = 1.0022E+00

91

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.11E-01	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
4.16E-02	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	SRY-EL
3.74E-02	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	SPY-E
3.59E-02	TC-S	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
3.31E-02	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-22	nCCI	SPY-E
3.31E-02	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-22	nCCI	SRY-EL
3.24E-02	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	SRY-EL
2.91E-02	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	SPY-E
2.56E-02	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-nn	RBB-nn	nCCI	SRY-EL
2.55E-02	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	nCCI	SRY-EL
2.29E-02	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	nCCI	SPY-E
2.01E-02	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
1.99E-02	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
1.98E-02	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	WCCI	SRY-EL
1.80E-02	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
1.79E-02	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	WCCI	SPY-E
1.56E-02	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.41E-02	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
1.17E-02	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	nCCI	SRY-EL
1.17E-02	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	WCCI	SRY-EL
1.11E-02	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
1.05E-02	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	WCCI	SPY-E
1.05E-02	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	nCCI	SPY-E
1.01E-02	TC-S	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n4	WCCI	SRY-EL
9.96E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	nCCI	SRY-EL
9.95E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
9.94E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	nCCI	SPY-E
9.31E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-44	nCCI	SPY-E
9.31E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-44	nCCI	SRY-EL
9.12E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	WCCI	SRY-EL
8.63E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
8.21E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	WCCI	SPY-E
7.77E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	WCCI	SRY-EL
7.76E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
7.75E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	WCCI	SPY-E
7.69E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
7.16E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	WCCI	SRY-EL
7.16E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	nCCI	SRY-EL
6.92E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	WCCI	SPY-E
6.64E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	nCCI	SRY-EL
6.63E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	nCCI	SPY-E
6.45E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	DCCI	SPY-E
6.44E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	nCCI	SPY-E
6.44E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	WCCI	SPY-E
5.58E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-24	WCCI	SRY-EL

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
5.30E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	nCCI	SRY-EL
5.18E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	WCCI	SRY-EL
5.16E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-22	WCCI	SPY-E
5.03E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-24	WCCI	SPY-E
4.77E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	nCCI	SPY-E
4.57E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	WCCI	SRY-EL
4.56E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	WCCI	SPY-E
4.43E-03	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-nn	RBB-nn	nCCI	SRY-EL
4.38E-03	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
4.14E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	WCCI	SRY-EL
3.93E-03	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
3.72E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	WCCI	SPY-E
3.54E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	DCCI	SPY-E
3.12E-03	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n3	nCCI	SRY-EL
3.06E-03	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	DCCI	SPY-E
2.80E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	nCCI	SRY-EL
2.80E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	WCCI	SRY-EL
2.80E-03	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n3	nCCI	SPY-E
2.80E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	nCCI	SPY-E
2.80E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-44	WCCI	SPY-E
2.46E-03	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	nCCI	SRY-EL
2.46E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
2.43E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
2.30E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	DCCI	SPY-E
2.22E-03	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	nCCI	SPY-E
2.22E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
2.19E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
2.19E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-24	WCCI	SRY-EL
2.18E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-24	WCCI	SPY-E
2.17E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
2.02E-03	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.95E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SPY-E
1.92E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.92E-03	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.92E-03	TC-S	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
1.90E-03	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	WCCI	SRY-EL
1.87E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	WCCI	SRY-EL
1.87E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	nCCI	SRY-EL
1.87E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	WCCI	SPY-E
1.87E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-44	nCCI	SPY-E
1.80E-03	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
1.80E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	DCCI	SPY-E
1.73E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
1.71E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
1.71E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	DCCI	SPY-E

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.70E-03	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-INM	VBD-DH	SPB-ny	RBB-n2	WCCI	SPY-E
1.70E-03	TC-S	CFD-n	CLC-nDD	CLV-022	CD-INM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
1.69E-03	TC-S	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
1.65E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	WCCI	SRY-EL
1.65E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	WCCI	SPY-E
1.54E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SPY-E
1.53E-03	TC-S	CFD-n	CLC-nDD	CLV-022	CD-INM	VBD-F	SPB-ny	RBB-n3	WCCI	SPY-E
1.50E-03	TC-S	CFD-n	CLC-nDD	CLV-022	CD-INM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.49E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-44	WCCI	SRY-EL
1.49E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-44	nCCI	SRY-EL
1.46E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-24	WCCI	SRY-EL
1.46E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-ny	RBB-24	WCCI	SPY-E
1.34E-03	TC-S	CFD-n	CLC-nDD	CLV-022	CD-INM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
1.33E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-44	nCCI	SPY-E
1.33E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-44	WCCI	SPY-E
1.29E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-24	WCCI	SRY-EL
1.29E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-24	WCCI	SPY-E
1.26E-03	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-INM	VBD-F	SPB-nn	RBB-44	DCCI	SPY-E
1.14E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-24	WCCI	SRY-EL
1.08E-03	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
1.05E-03	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-22	nCCI	SRY-EL
1.03E-03	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-INM	VBD-F	SPB-nn	RBB-n2	nCCI	SRY-EL
1.03E-03	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-24	WCCI	SPY-E
9.85E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-INM	VBD-F	SPB-nn	RBB-24	DCCI	SPY-E
9.76E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SPY-E
9.66E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-S	SPB-ny	RBB-22	WCCI	SRY-EL
9.41E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-22	nCCI	SPY-E
9.31E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-INM	VBD-F	SPB-nn	RBB-n2	nCCI	SPY-E
8.86E-04	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
8.69E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-S	SPB-ny	RBB-22	WCCI	SPY-E
8.46E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	DCCI	SPY-E
8.36E-04	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-INM	VBD-F	SPB-ny	RBB-n4	DCCI	SPY-E
8.16E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-22	WCCI	SRY-EL
8.10E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-22	DCCI	SPY-E
8.07E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-INM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
8.00E-04	TC-S	CFD-n	CLC-nnW	CLV-002	CD-ILH	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
7.34E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-INM	VBD-DH	SPB-ny	RBB-22	WCCI	SPY-E
7.26E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-INM	VBD-F	SPB-nn	RBB-n2	WCCI	SPY-E
6.86E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-INM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
6.39E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	nCCI	SPY-E
6.39E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-22	nCCI	SRY-EL
6.17E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-INM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
5.41E-04	TC-S	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
5.37E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
5.35E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-INM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
5.16E-04	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n4	WCCI	SRY-EL
4.99E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-22	WCCI	SRY-EL
4.99E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-22	WCCI	SPY-E
4.83E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
4.82E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n4	DCCI	SPY-E
4.82E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
4.80E-04	TC-S	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
4.78E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
4.73E-04	TC-S	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-S	SPB-nn	RBB-n4	WCCI	SRY-EL
4.66E-04	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n4	WCCI	SPY-E
4.60E-04	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
4.31E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	WCCI	SPY-E
4.14E-04	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
3.81E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	DCCI	SPY-E
3.70E-04	TC-S	CFD-E4V	CLC-WWD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	DCCI	SPY-E
3.67E-04	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n2	DCCI	SPY-E
3.67E-04	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
3.49E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-44	WCCI	SRY-EL
3.40E-04	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
3.30E-04	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB n2	WCCI	SPY-E
3.20E-04	TC-S	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
3.19E-04	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n3	DCCI	SPY-E
3.14E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-44	WCCI	SPY-E
3.06E-04	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
2.93E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	DCCI	SPY-E
2.87E-04	TC-S	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-pvn	RBB-n2	WCCI	SRY-EL
2.79E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	nCCI	SRY-EL
2.79E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	WCCI	SRY-EL
2.73E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	nCCI	SRY-EL
2.72E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-24	WCCI	SRY-EL
2.61E-04	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n4	DCCI	SPY-E
2.61E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-pvn	RBB-22	nCCI	SRY-EL
2.61E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-pvn	RBB-22	nCCI	SPY-E
2.51E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	nCCI	SPY-E
2.51E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	WCCI	SPY-E
2.46E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	nCCI	SPY-E
2.45E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-24	WCCI	SPY-E
2.42E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
2.41E-04	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
2.39E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n4	DCCI	SPY-E
2.35E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	WCCI	SRY-EL
2.29E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	DCCI	SPY-E
2.28E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-nn	RBB-n4	WCCI	SRY-EL
2.26E-04	TC-S	CFD-n	CLC-nnW	CLV-002	CD-ILM	VBD-F	SPB-nn	RBB-n4	WCCI	SRY-EL
2.17E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-ny	RBB-n2	nCCI	SPY-E

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
2.13E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	WCCI	SRY-EL
2.11E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	WCCI	SPY-E
2.05E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-nn	RBB-n4	WCCI	SPY-E
2.02E-04	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.97E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SRY-EL
1.92E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	WCCI	SPY-E
1.86E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	DCCI	SPY-E
1.80E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-44	WCCI	SRY-EL
1.80E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-44	WCCI	SPY-E
1.80E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	nCCI	SPY-E
1.80E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-ny	RBB-44	nCCI	SRY-EL
1.77E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPY-E
1.59E-04	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
1.44E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SRY-EL
1.44E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.41E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-24	WCCI	SRY-EL
1.41E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-24	WCCI	SPY-E
1.39E-04	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	DCCI	SPY-E
1.38E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-nn	RBB-n2	DCCI	SPY-E
1.38E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	nCCI	SRY-EL
1.38E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SPY-E
1.38E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SRY-EL
1.35E-04	TC-S	CFD-LP	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-nn	RBB-n4	WCCI	SRY-EL
1.30E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPY-E
1.29E-04	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
1.24E-04	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	nCCI	SPY-E
1.20E-04	TC-S	CFD-E4V	CLC-WWD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	DCCI	SPY-E
1.19E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SRY-EL
1.10E-04	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
1.09E-04	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-nn	RBB-n2	nCCI	SRY-EL
1.07E-04	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n4	DCCI	SPY-E
1.07E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	WCCI	SRY-EL
1.07E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	WCCI	SPY-E
1.07E-04	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPY-E
9.80E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-nn	RBB-n2	nCCI	SPY-E
9.60E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
9.38E-05	TC-S	CFD-E4V	CLC-WWD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	DCCI	SPY-E
9.05E-05	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
9.05E-05	TC-S	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
8.86E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
8.69E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	DCCI	SPY-E
8.64E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n4	WCCI	SPY-E
8.62E-05	TC-S	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
8.62E-05	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
8.49E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL

96

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
8.36E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SRY-EL
8.26E-05	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n2	DCCI	SPY-E
8.25E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
8.25E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SRY-EL
8.25E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	nCCI	SPY-E
8.23E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
8.12E-05	TC-S	CFD-LP	CLC-nnW	CLV-003	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
8.10E-05	TC-S	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-pvn	RBB-n4	WCCI	SRY-EL
7.97E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
7.97E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	nCCI	SPY-E
7.73E-05	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-S	SPB-ny	RBB-n3	WCCI	SRY-EL
7.64E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-nn	RBB-n2	WCCI	SPY-E
7.52E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPY-E
7.43E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SPY-E
7.42E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-22	nCCI	SRY-EL
7.41E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E
7.37E-05	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
7.18E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SPY-E
7.04E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	nCCI	SPY-E
7.04E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	nCCI	SRY-EL
7.00E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-DH	SPB-ny	RBB-n4	DCCI	SPY-E
6.96E-05	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILH	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
6.94E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-44	nCCI	SRY-EL
6.94E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-44	WCCI	SRY-EL
6.91E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
6.80E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	DCCI	SPY-E
6.68E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-22	nCCI	SPY-E
6.63E-05	TC-S	CFD-n	CLC-nnD	CLV-002	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
6.62E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-44	nCCI	SRY-EL
6.62E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-pvn	RBB-44	nCCI	SPY-E
6.53E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SRY-EL
6.44E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
6.44E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	WCCI	SPY-E
6.44E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-22	WCCI	SRY-EL
6.42E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
6.39E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-22	nCCI	SPY-E
6.39E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-22	nCCI	SRY-EL
6.25E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-44	nCCI	SPY-E
6.25E-05	TC-S	CFD-E4V	[illegible]-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-44	WCCI	SPY-E
6.22E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
6.22E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n2	WCCI	SPY-E
6.15E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
5.92E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-22	nCCI	SPY-E
5.92E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-22	nCCI	SRY-EL
5.88E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	nCCI	SPY-E

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
5.87E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	SPY-E
5.87E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-22	nCCI	SRY-EL
5.79E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SPY-E
5.79E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-22	WCCI	SRY-EL
5.78E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
5.73E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-22	WCCI	SRY-EL
5.73E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-22	WCCI	SPY-E
5.60E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SPY-E
5.54E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n2	WCCI	SPY-E
5.50E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	WCCI	SPY-E
5.50E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-22	WCCI	SRY-EL
5.41E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-24	WCCI	SRY-EL
5.37E-05	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-S	SPB-nn	RBB-nn	nCCI	SRY-EL
5.21E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-22	WCCI	SPY-E
5.21E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SRY-EL
5.21E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SRY-EL
5.14E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
5.10E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SRY-EL
5.09E-05	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
5.05E-05	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-vn	RBB-nn	nCCI	SRY-EL
5.02E-05	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	DCCI	SPY-E
4.91E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SRY-EL
4.87E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-24	WCCI	SPY-E
4.78E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n4	DCCI	SPY-E
4.70E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	nCCI	SPY-E
4.70E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	nCCI	SRY-EL
4.69E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPY-E
4.69E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPY-E
4.62E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	nCCI	SPY-E
4.59E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-ny	RBB-n2	WCCI	SPY-E
4.57E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-S	SPB-nn	RBB-22	WCCI	SRY-EL
4.42E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	nCCI	SPY-E
4.31E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-pvn	RBB-22	nCCI	SRY-EL
4.21E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n2	nCCI	SRY-EL
4.12E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-S	SPB-nn	RBB-22	WCCI	SPY-E
4.07E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	SPRY-L
4.06E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SRY-EL
4.01E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
3.92E-05	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	DCCI	SPY-E
3.88E-05	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-nn	RBB-n4	DCCI	SPY-E
3.88E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-pvn	RBB-22	nCCI	SPY-E
3.83E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SRY-EL
3.79E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n2	nCCI	SPY-E
3.66E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	WCCI	SRY-EL
3.66E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-22	WCCI	SPY-E

98

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
3.66E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	nSPY
3.65E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPY-E
3.61E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F-B	SPB-ny	RBB-n2	WCCI	SPY-E
3.55E-05	TC-S	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SPRY-L
3.55E-05	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-F	SPB-pvn	RBB-nn	nCCI	SRY-EL
3.49E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-44	nCCI	SRY-EL
3.49E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	nCCI	SPY-E
3.49E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	nCCI	SRY-EL
3.49E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	WCCI	SPY-E
3.49E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	WCCI	SRY-EL
3.45E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	DCCI	SPY-E
3.43E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-SE	SPB-ny	RBB-n2	WCCI	SPY-E
3.36E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-pvn	RBB-22	WCCI	SRY-EL
3.35E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-DH	SPB-ny	RBB-n3	nCCI	SRY-EL
3.26E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	WCCI	SPY-E
3.26E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-22	WCCI	SRY-EL
3.17E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	SPRY-L
3.16E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SRY-EL
3.14E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-44	nCCI	SPY-E
3.14E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-44	DCCI	SPY-E
3.06E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
3.02E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-pvn	RBB-22	WCCI	SPY-E
3.02E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SRY-EL
3.02E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SRY-EL
3.02E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-DH	SPB-ny	RBB-n3	nCCI	SPY-E
2.96E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-S	SPB-ny	RBB-n2	DCCI	SPY-E
2.94E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-22	nCCI	nSPY
2.94E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-nn	RBB-22	nCCI	SPRY-L
2.86E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	nSPY
2.84E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-ny	RBB-n2	WCCI	SPY-E
2.80E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	nCCI	SRY-EL
2.76E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILM	VBD-S-B	SPB-ny	RBB-n2	WCCI	SPY-E
2.73E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F	SPB-ny	RBB-n3	nCCI	SRY-EL
2.72E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-24	WCCI	SRY-EL
2.72E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-24	WCCI	SPY-E
2.72E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPY-E
2.72E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPY-E
2.62E-05	TC-S	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SRY-EL
2.62E-05	TC-S	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SRY-EL
2.55E-05	TC-S	CFD-LP	CLC-nnD	CLV-003	CD-ILH	VBD-S	SPB-ny	RBB-n4	WCCI	SRY-EL
2.52E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	nCCI	SPY-E
2.50E-05	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILH	VBD-nB	SPB-nn	RBB-nn	nCCI	SPRY-L
2.47E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-nn	RBB-22	DCCI	SPY-E
2.46E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F	SPB-ny	RBB-n3	nCCI	SPY-E
2.45E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-24	DCCI	SPY-E

Table A-6
Central Estimate Detailed Results For The TC5X Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
2.43E-05	TC-S	CFD-LP	CLC-nnD	CLV-003	CD-ILM	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
2.40E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-nn	RBB-n4	WCCI	SRY-EL
2.36E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SRY-EL
2.36E-05	TC-S	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	WCCI	SPY-E
2.36E-05	TC-S	CFD-LV	CLC-nDW	CLV-023	CD-ILM	VBD-F-B	SPB-ny	RBB-n2	nCCI	SPY-E
2.33E-05	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-F	SPB-ny	RBB-n4	DCCI	SPY-E
2.33E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	WCCI	SPY-E
2.33E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	nCCI	SRY-EL
2.33E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	nCCI	SPY-E
2.33E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-44	WCCI	SRY-EL
2.26E-05	TC-S	CFD-LP	CLC-nnW	CLV-003	CD-ILH	VBD-S	SPB-nn	RBB-n2	WCCI	SRY-EL
2.18E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	WCCI	SRY-EL
2.16E-05	TC-S	CFD-n	CLC-nWW	CLV-022	CD-IHM	VBD-F	SPB-nn	RBB-n4	WCCI	SPY-E
2.15E-05	TC-S	CFD-LP	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n2	WCCI	SRY-EL
2.12E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPY-E
2.07E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-44	WCCI	SPY-E
2.07E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-44	WCCI	SRY-EL
1.99E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	WCCI	SPY-E
1.99E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	WCCI	SRY-EL
1.99E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	nCCI	SPY-E
1.99E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-44	nCCI	SRY-EL
1.97E-05	TC-S	CFD-n	CLC-nnn	CLV-000	CD-ILM	VBD-F	SPB-nn	RBB-nn	nCCI	SPRY-L
1.97E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.97E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S-B	SPB-ny	RBB-n2	WCCI	SRY-EL
1.96E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	WCCI	SPY-E
1.95E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	SRY-EL
1.87E-05	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-23	nCCI	SRY-EL
1.82E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-24	WCCI	SPY-E
1.82E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F-B	SPB-ny	RBB-24	WCCI	SRY-EL
1.80E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-44	nCCI	SPY-E
1.80E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-S	SPB-nn	RBB-44	nCCI	SRY-EL
1.80E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	nCCI	SPRY-L
1.78E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	nCCI	SRY-EL
1.78E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	nCCI	SPY-E
1.78E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	nCCI	SPY-E
1.77E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-S-B	SPB-ny	RBB-n2	WCCI	SPY-E
1.76E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	SPY-E
1.76E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	nCCI	SRY-EL
1.75E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-pvy	RBB-n4	WCCI	SPY-E
1.74E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	SRY-EL
1.70E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F-B	SPB-ny	RBB-n3	nCCI	SRY-EL
1.69E-05	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-23	nCCI	SPY-E
1.68E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-DH	SPB-ny	RBB-n3	WCCI	SRY-EL
1.67E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-44	nCCI	SRY-EL
1.67E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-vn	RBB-44	nCCI	SPY-E

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.66E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-44	nCCI	SPY-E
1.66E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-nB	SPB-vn	RBB-44	nCCI	SRY-EL
1.65E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-S	SPB-nn	RBB-44	WCCI	SRY-EL
1.62E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-24	WCCI	SRY-EL
1.62E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S-B	SPB-ny	RBB-24	WCCI	SPY-E
1.62E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	nCCI	nSPY
1.56E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-S	SPB-pvy	RBB-n4	WCCI	SPY-E
1.55E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-24	WCCI	SRY-EL
1.55E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-pvn	RBB-24	WCCI	SPY-E
1.55E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-pvy	RBB-n2	nCCI	SRY-EL
1.54E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.53E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F-B	SPB-ny	RBB-n3	nCCI	SPY-E
1.51E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-DH	SPB-ny	RBB-n4	WCCI	SRY-EL
1.51E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-DH	SPB-ny	RBB-n3	WCCI	SPY-E
1.49E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-S	SPB-nn	RBB-44	WCCI	SPY-E
1.46E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F-B	SPB-ny	RBB-n3	nCCI	SRY-EL
1.45E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	nCCI	SPY-E
1.45E-05	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	nCCI	SPY-E
1.45E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	nCCI	SRY-EL
1.45E-05	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	nCCI	SRY-EL
1.40E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	WCCI	SPRY-L
1.39E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-pvy	RBB-n2	nCCI	SPY-E
1.39E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	WCCI	SRY-EL
1.39E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-vn	RBB-22	WCCI	SPY-E
1.38E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-pvy	RBB-n2	WCCI	SPY-E
1.37E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	WCCI	SRY-EL
1.37E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-ILH	VBD-F	SPB-vn	RBB-22	WCCI	SPY-E
1.37E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.36E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F	SPB-ny	RBB-n3	WCCI	SRY-EL
1.36E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-DH	SPB-ny	RBB-n4	WCCI	SPY-E
1.35E-05	TC-S	CFD-LV	CLC-nWW	CLV-003	CD-ILM	VBD-S	SPB-pvn	RBB-n2	WCCI	SRY-EL
1.32E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	WCCI	SPY-E
1.32E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	nCCI	SPY-E
1.32E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	WCCI	SRY-EL
1.32E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-44	nCCI	SRY-EL
1.31E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-IHM	VBD-F-B	SPB-ny	RBB-n3	nCCI	SPY-E
1.29E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-S	SPB-nn	RBB-24	WCCI	SRY-EL
1.28E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-44	WCCI	SRY-EL
1.28E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-44	nCCI	SRY-EL
1.26E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	WCCI	nSPY
1.23E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILH	VBD-S	SPB-pvy	RBB-n2	WCCI	SPY-E
1.23E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F	SPB-ny	RBB-n3	WCCI	SPY-E
1.23E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL
1.22E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-pvn	RBB-44	nCCI	SRY-EL
1.22E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-pvn	RBB-44	WCCI	SRY-EL

Table A-6
Central Estimate Detailed Results For The TCSX Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.21E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-pvy	RBB-n2	WCCI	SRY-EL
1.18E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	nCCI	SRY-EL
1.18E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-yy	RBB-22	nCCI	SPY-E
1.18E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-44	WCCI	SPY-E
1.18E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-S	SPB-pvy	RBB-44	WCCI	SRY-EL
1.17E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	nCCI	SPY-E
1.17E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-yy	RBB-22	nCCI	SRY-EL
1.16E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-S	SPB-nn	RBB-24	WCCI	SPY-E
1.15E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-44	WCCI	SPY-E
1.15E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-44	nCCI	SPY-E
1.13E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	WCCI	SRY-EL
1.13E-05	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	WCCI	SPY-E
1.13E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	WCCI	SPY-E
1.13E-05	TC-S	CFD-IP	CLC-WDD	CLV-333	CD-ILH	VBD-SE	SPB-ny	RBB-22	WCCI	SRY-EL
1.13E-05	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-DH	SPB-nn	RBB-n2	nCCI	SRY-EL
1.11E-05	TC-S	CFD-IM	CLC-nDD	CLV-033	CD-ILM	VBD-F	SPB-ny	RBB-n2	nCCI	SPRY-L
1.10E-05	TC-S	CFD-n	CLC-nDD	CLV-022	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SPY-E
1.09E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-pvn	RBB-44	nCCI	SPY-E
1.09E-05	TC-S	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-pvn	RBB-44	WCCI	SPY-E
1.09E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-IHM	VBD-F	SPB-pvy	RBB-n2	WCCI	SPY-E
1.03E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-24	WCCI	SRY-EL
1.03E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-ILH	VBD-F	SPB-pvy	RBB-24	WCCI	SPY-E
1.02E-05	TC-S	CFD-IP	CLC-nWW	CLV-033	CD-IHM	VBD-DH	SPB-nn	RBB-n2	nCCI	SPY-E
1.00E-05	TC-S	CFD-LV	CLC-nnW	CLV-003	CD-ILM	VBD-F	SPB-nn	RBB-n4	WCCI	SPRY-L
1.00E-05	TC-S	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-pvy	RBB-24	WCCI	SRY-EL
1.00E-05	TC-S	CFD-IP	CLC-nDD	CLV-033	CD-ILH	VBD-F	SPB-ny	RBB-n4	WCCI	SRY-EL

9.9227E-01 TOTAL FREQUENCY OF BINS PRINTED

TOTAL FREQUENCY IN ALL BINS = 1.0000E+00
TOTAL FREQUENCY LOST DUE TO TERMINATION OF BRANCHES 7.7332E-03
TOTAL FREQUENCY FOR THIS SEQUENCE/WALKTHROUGH = 1.0077E+00

Table A-7
Central Estimate Detailed Results For The TCSRX2 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
5.576E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	SPRY-L
5.548E-02	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-32	nCCI	SPRY-L
5.019E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	nSPY
4.992E-02	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-32	nCCI	nSPY
4.348E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	SPRY-L
4.325E-02	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-32	WCCI	SPRY-L
3.913E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	nSPY
3.893E-02	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-32	WCCI	nSPY
3.002E-02	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	nCCI	SPRY-L
2.984E-02	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-33	nCCI	SPRY-L
2.700E-02	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	nCCI	nSPY
2.686E-02	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-33	nCCI	nSPY
2.339E-02	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	WCCI	SPRY-L
2.105E-02	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	WCCI	nSPY
1.626E-02	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-33	WCCI	SPRY-L
1.571E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	nCCI	SPRY-L
1.571E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	WCCI	SPRY-L
1.463E-02	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-33	WCCI	nSPY
1.413E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	WCCI	nSPY
1.413E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	nCCI	nSPY
1.356E-02	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-34	WCCI	SPRY-L
1.223E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	WCCI	SPRY-L
1.220E-02	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-34	WCCI	nSPY
1.218E-02	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-34	WCCI	SPRY-L
1.101E-02	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	WCCI	nSPY
1.097E-02	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-34	WCCI	nSPY
1.059E-02	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
9.512E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
8.651E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	DCCI	nSPY
8.594E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-32	DCCI	nSPY
8.446E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	nCCI	SPRY-L
8.446E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	WCCI	SPRY-L
8.243E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
8.076E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-32	nCCI	SPRY-L
7.592E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	nCCI	nSPY
7.592E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	WCCI	nSPY
7.419E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
7.268E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-32	nCCI	nSPY
6.779E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	nCCI	SPRY-L
6.571E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-24	WCCI	SPRY-L
6.299E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-32	WCCI	SPRY-L
6.101E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	nCCI	nSPY
5.913E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-24	WCCI	nSPY
5.669E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-32	WCCI	nSPY
5.460E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-42	nCCI	SPRY-L

Table A-7
Central Estimate Detailed Results For The TCSRX2 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
5.288E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	WCCI	SPRY-L
4.914E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-42	nCCI	nSPY
4.759E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	WCCI	nSPY
4.258E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-42	WCCI	SPRY-L
4.146E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	DCCI	nSPY
3.633E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-42	WCCI	nSPY
3.217E-03	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-33	DCCI	nSPY
3.064E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	DCCI	nSPY
2.981E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
2.981E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
2.940E-03	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
2.940E-03	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
2.701E-03	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-34	DCCI	nSPY
2.683E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
2.683E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
2.646E-03	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
2.646E-03	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
2.402E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-34	DCCI	nSPY
2.355E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	DCCI	nSPY
2.325E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
2.093E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
1.903E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	WCCI	SPRY-L
1.903E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	nCCI	SPRY-L
1.736E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-34	WCCI	SPRY-L
1.713E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	nCCI	nSPY
1.713E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	WCCI	nSPY
1.599E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	DCCI	nSPY
1.562E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-34	WCCI	nSPY
1.518E-03	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	nCCI	SPRY-L
1.497E-03	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	SPRY-L
1.466E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	WCCI	SPRY-L
1.446E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	DCCI	nSPY
1.348E-03	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	nCCI	nSPY
1.347E-03	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	nSPY
1.320E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	WCCI	nSPY
1.239E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	WCCI	SPRY-L
1.232E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-32	WCCI	SPRY-L
1.231E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-32	DCCI	nSPY
1.201E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
1.186E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.169E-03	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	WCCI	SPRY-L
1.167E-03	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	SPRY-L
1.128E-03	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-24	DCCI	nSPY
1.115E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	WCCI	nSPY
1.109E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-32	WCCI	nSPY

Table A-7
Central Estimate Detailed Results For The TCSRX2 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.081E-03	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
1.068E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
1.052E-03	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	WCCI	nSPY
1.051E-03	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	nSPY
1.042E-03	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.014E-03	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	DCCI	nSPY
9.765E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
9.378E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	nCCI	nSPY
9.254E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
8.806E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
8.517E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-42	DCCI	nSPY
8.329E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
8.127E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	WCCI	SPRY-L
7.891E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-42	nCCI	SPRY-L
7.632E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
7.315E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	WCCI	nSPY
7.102E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-42	nCCI	nSPY
6.677E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
6.279E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
6.155E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-42	WCCI	SPRY-L
5.651E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
5.544E-04	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-F-B	SPB-ny	RBB-44	DCCI	nSPY
5.539E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-42	WCCI	nSPY
5.411E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
5.269E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	DCCI	nSPY
4.942E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	nCCI	SPRY-L
4.870E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
4.698E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
4.448E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	nCCI	nSPY
4.367E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VDD-S	SPB-ny	RBB-44	WCCI	SPRY-L
4.228E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
4.153E-04	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-33	nCCI	SPRY-L
4.110E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	DCCI	nSPY
4.068E-04	TC-V	CFD-E4V	CLC-WWD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	DCCI	nSPY
4.021E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
3.930E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VDD-S	SPB-ny	RBB-44	WCCI	nSPY
3.870E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	nCCI	SPRY-L
3.870E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	WCCI	SPRY-L
3.738E-04	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-33	nCCI	nSPY
3.709E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	nCCI	SPRY-L
3.709E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	WCCI	SPRY-L
3.655E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	WCCI	SPRY-L
3.619E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
3.483E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	WCCI	nSPY
3.483E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	nCCI	nSPY

Table A-7
Central Estimate Detailed Results For The TCSRX2 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
3.475E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-34	WCCI	SPRY-L
3.472E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-34	DCCI	nSPY
3.419E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	nCCI	SPRY-L
3.406E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-24	WCCI	SPRY-L
3.341E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	DCCI	nSPY
3.290E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	WCCI	nSPY
3.128E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-34	WCCI	nSPY
3.093E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	SPRY-L
3.077E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	nCCI	nSPY
3.066E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-24	WCCI	nSPY
3.058E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	nCCI	nSPY
3.058E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	WCCI	nSPY
3.018E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	WCCI	SPRY-L
2.806E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	SPRY-L
2.806E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	nCCI	SPRY-L
2.784E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	nCCI	nSPY
2.737E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
2.737E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
2.716E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	WCCI	nSPY
2.667E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	WCCI	SPRY-L
2.651E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	WCCI	SPRY-L
2.525E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	nSPY
2.525E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	nCCI	nSPY
2.463E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
2.463E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
2.402E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-F-B	SPB-ny	RBB-44	DCCI	nSPY
2.400E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-22	WCCI	nSPY
2.275E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	DCCI	nSPY
2.257E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
2.257E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
2.213E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	SPRY-L
2.193E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	WCCI	nSPY
2.189E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-24	WCCI	SPRY-L
2.077E-04	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-33	WCCI	SPRY-L
2.048E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
1.992E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	WCCI	nSPY
1.934E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
1.934E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
1.869E-04	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-33	WCCI	nSPY
1.869E-04	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-34	WCCI	SPRY-L
1.773E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-24	WCCI	nSPY
1.755E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-32	nCCI	SPRY-L
1.736E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	SPRY-L
1.728E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-22	DCCI	nSPY
1.682E-04	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-34	WCCI	nSPY

Table A-7
Central Estimate Detailed Results For The TCSRX2 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.676E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
1.645E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
1.580E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-32	nCCI	nSPY
1.562E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	nSPY
1.536E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-22	DCCI	nSPY
1.448E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.435E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
1.435E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
1.398E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-S-B	SPB-ny	RBB-22	WCCI	SPRY-L
1.397E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	DCCI	nSPY
1.369E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	nCCI	SPRY-L
1.316E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
1.303E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	nCCI	nSPY
1.296E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	DCCI	nSPY
1.291E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
1.291E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
1.258E-04	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-S-B	SPB-ny	RBB-22	WCCI	nSPY
1.232E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-42	WCCI	SPRY-L
1.232E-04	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	nCCI	nSPY
1.232E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-32	DCCI	nSPY
1.220E-04	TC-V	CFD-E4V	CLC-WWD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	DCCI	nSPY
1.201E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.190E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
1.190E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	WCCI	SPRY-L
1.190E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
1.190E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	nCCI	SPRY-L
1.109E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-42	WCCI	nSPY
1.081E-04	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	nCCI	nSPY
1.071E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
1.071E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	WCCI	nSPY
1.071E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
1.071E-04	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-44	nCCI	nSPY
1.034E-04	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-DH-B	SPB-ny	RBB-42	DCCI	nSPY
1.032E-04	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
9.611E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	DCCI	nSPY
9.513E-05	TC-V	CFD-E4V	CLC-WWD	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	DCCI	nSPY
9.366E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	WCCI	SPRY-L
9.279E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-24	WCCI	SPRY-L
9.279E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
8.454E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	DCCI	nSPY
8.429E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	WCCI	nSPY
8.351E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
8.351E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
8.351E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-24	WCCI	nSPY
7.675E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-44	WCCI	SPRY-L

Table A-7
Central Estimate Detailed Results For The TCSRX2 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
7.675E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-44	nCCI	SPRY-L
7.493E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	WCCI	SPRY-L
7.138E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	WCCI	SPRY-L
7.138E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	nCCI	SPRY-L
6.907E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-44	nCCI	nSPY
6.907E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-44	WCCI	nSPY
6.743E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-22	WCCI	nSPY
6.424E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	nCCI	nSPY
6.424E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-44	WCCI	nSPY
5.966E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-24	WCCI	SPRY-L
5.922E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	DCCI	nSPY
5.922E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-F-B	SPB-ny	RBB-22	DCCI	nSPY
5.922E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-nn	RBB-22	DCCI	nSPY
5.568E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	WCCI	SPRY-L
5.555E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-44	DCCI	nSPY
5.388E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-DH	SPB-nn	RBB-24	WCCI	nSPY
5.059E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-S	SPB-nn	RBB-22	WCCI	SPRY-L
5.056E-05	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-S-B	SPB-ny	RBB-44	WCCI	SPRY-L
5.051E-05	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	DCCI	nSPY
5.011E-05	TC-V	CFD-IP	CLC-WWW	CLV-333	CD-IHM	VBD-F	SPB-ny	RBB-24	WCCI	nSPY
4.553E-05	TC-V	CFD-E4V	CLC-WWW	CLV-333	CD-IHM	VBD-S	SPB-nn	RBB-22	WCCI	nSPY
4.550E-05	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-S-B	SPB-ny	RBB-44	WCCI	nSPY
4.333E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-24	DCCI	nSPY
4.157E-05	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-S-B	SPB-ny	RBB-33	WCCI	SPRY-L
3.944E-05	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-S-B	SPB-ny	RBB-24	WCCI	SPRY-L
3.862E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-44	nCCI	SPRY-L
3.742E-05	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-S-B	SPB-ny	RBB-33	WCCI	nSPY
3.742E-05	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-S-B	SPB-ny	RBB-34	WCCI	SPRY-L
3.549E-05	TC-V	CFD-IP	CLC-DDD	CLV-333	CD-IHM	VBD-S-B	SPB-ny	RBB-24	WCCI	nSPY
3.489E-05	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	SPRY-L
3.489E-05	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	nCCI	SPRY-L
3.475E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-44	nCCI	nSPY
3.475E-05	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-44	WCCI	SPRY-L
3.475E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-44	DCCI	nSPY
3.475E-05	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-34	DCCI	nSPY
3.387E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	nCCI	SPRY-L
3.387E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	SPRY-L
3.367E-05	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-S-B	SPB-ny	RBB-34	WCCI	nSPY
3.140E-05	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	nSPY
3.140E-05	TC-V	CFD-n	CLC-DDD	CLV-222	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	nCCI	nSPY
3.128E-05	TC-V	CFD-IM	CLC-DDD	CLV-233	CD-IHM	VBD-S-B	SPB-ny	RBB-44	WCCI	nSPY
3.046E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	nSPY
3.048E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	nCCI	nSPY
2.711E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-24	DCCI	nSPY
2.709E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	SPRY-L

Table A-7
Central Estimate Detailed Results For The TCSRX2 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
2.709E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	nCCI	SPRY-L
2.642E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-24	WCCI	SPRY-L
2.500E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	WCCI	SPRY-L
2.438E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	nCCI	nSPY
2.438E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-44	WCCI	nSPY
2.378E-05	TC-V	CFD-E4V	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-24	WCCI	nSPY
2.250E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-S	SPB-ny	RBB-22	WCCI	nSPY
2.113E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH-B	SPB-ny	RBB-24	WCCI	SPRY-L
2.073E-05	TC-V	CFD-IP	CLC-WDD	CLV-333	CD-IHM	VBD-DH	SPB-ny	RBB-23	nCCI	SPRY-L

9.8929E-01 TOTAL FREQUENCY OF BINS PRINTED

TOTAL FREQUENCY IN ALL BINS = 1.0000E+00
TOTAL FREQUENCY LOST DUE TO TERMINATION OF BRANCHES 1.0709E-02
TOTAL FREQUENCY FOR THIS SEQUENCE/WALKTHROUGH = 1.0107E+00

Table A-8
Central Estimate Detailed Results For The TCSRV23 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
6.422E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-nB	SPB-nn	RBB-22	nCCI	SPRY-L
6.422E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-nB	SPB-nn	RBB-22	nCCI	nSPY
6.417E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
5.775E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
5.005E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
4.502E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
3.206E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
3.206E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
2.501E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
2.501E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
2.255E-02	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-nn	RBB-22	nCCI	nSPY
2.255E-02	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-nn	RBB-22	nCCI	SPRY-L
1.809E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-nB	SPB-nn	RBB-44	nCCI	nSPY
1.809E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
1.809E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-nB	SPB-nn	RBB-44	nCCI	SPRY-L
1.809E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
1.628E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
1.628E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
1.410E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
1.269E-02	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
9.970E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	DCCI	nSPY
9.899E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	WCCI	SPRY-L
9.899E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	WCCI	SPRY-L
9.899E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	WCCI	nSPY
9.013E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
9.013E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
9.013E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
9.013E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
8.909E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	WCCI	nSPY
7.753E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-22	nCCI	SPRY-L
7.030E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
7.030E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
6.979E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-22	nCCI	nSPY
6.361E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-nn	RBB-44	nCCI	nSPY
6.361E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-nn	RBB-44	nCCI	SPRY-L
6.343E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-22	nCCI	SPRY-L
6.047E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-22	WCCI	SPRY-L
5.709E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-22	nCCI	nSPY
5.443E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-22	WCCI	nSPY
5.169E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
4.948E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-22	WCCI	SPRY-L
4.652E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
4.453E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-22	WCCI	nSPY
4.229E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-22	nCCI	nSPY
4.229E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-22	nCCI	SPRY-L

Table A-8
Central Estimate Detailed Results For The TCSRV23 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
4.229E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
4.032E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
3.806E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
3.628E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
3.605E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	DCCI	nSPY
3.569E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	WCCI	SPRY-L
3.569E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	WCCI	SPRY-L
3.569E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	WCCI	nSPY
3.299E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-22	WCCI	SPRY-L
3.299E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-22	WCCI	nSPY
3.299E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
3.212E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	WCCI	nSPY
2.969E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
2.819E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-22	nCCI	SPRY-L
2.819E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
2.819E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
2.819E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-22	nCCI	nSPY
2.812E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	DCCI	nSPY
2.784E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-24	WCCI	nSPY
2.784E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-24	WCCI	SPRY-L
2.784E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-24	WCCI	SPRY-L
2.505E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-24	WCCI	nSPY
2.199E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-22	WCCI	nSPY
2.199E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-22	WCCI	SPRY-L
2.187E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-44	WCCI	SPRY-L
2.187E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-44	nCCI	SPRY-L
2.179E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
2.179E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
1.968E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-44	nCCI	nSPY
1.968E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-44	WCCI	nSPY
1.959E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-22	WCCI	nSPY
1.959E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-22	WCCI	SPRY-L
1.959E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-22	WCCI	SPRY-L
1.844E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.844E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	nCCI	nSPY
1.789E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-44	WCCI	SPRY-L
1.789E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-44	nCCI	SPRY-L
1.763E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-22	WCCI	nSPY
1.706E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-24	WCCI	SPRY-L
1.610E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-44	WCCI	nSPY
1.610E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-44	nCCI	nSPY
1.535E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-24	WCCI	nSPY
1.438E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	WCCI	nSPY
1.438E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-22	WCCI	SPRY-L
1.430E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L

Table A-8
Central Estimate Detailed Results For The TCSRV23 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.430E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
1.396E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-24	WCCI	SPRY-L
1.379E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.379E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.379E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	nCCI	nSPY
1.287E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
1.287E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
1.281E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	WCCI	SPRY-L
1.281E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	WCCI	nSPY
1.281E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	WCCI	SPRY-L
1.256E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-24	WCCI	nSPY
1.241E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	nCCI	nSPY
1.193E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-44	WCCI	nSPY
1.193E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-44	WCCI	SPRY-L
1.193E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-44	nCCI	nSPY
1.193E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-44	nCCI	SPRY-L
1.170E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
1.170E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
1.153E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	WCCI	nSPY
1.116E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
1.089E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-22	DCCI	nSPY
1.053E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
1.053E-03	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
1.048E-03	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	DCCI	nSPY
1.004E-03	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
9.896E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-22	DCCI	nSPY
9.397E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-ny	RBB-22	nCCI	SPRY-L
9.397E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-ny	RBB-22	nCCI	nSPY
9.303E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-24	WCCI	nSPY
9.303E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F	SPB-nn	RBB-24	WCCI	SPRY-L
9.127E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
8.214E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
7.952E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-44	WCCI	SPRY-L
7.952E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-44	WCCI	nSPY
7.952E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-44	nCCI	SPRY-L
7.952E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-44	nCCI	nSPY
7.801E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
7.801E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
7.801E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
7.801E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
7.707E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	DCCI	nSPY
7.084E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-44	WCCI	SPRY-L
7.084E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-44	WCCI	SPRY-L
7.084E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-44	WCCI	nSPY
6.472E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-22	DCCI	nSPY

Table A-8
Central Estimate Detailed Results For The TCSRV23 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
6.376E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-44	WCCI	nSPY
6.202E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-24	WCCI	nSPY
6.202E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F	SPB-ny	RBB-24	WCCI	SPRY-L
6.084E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
6.084E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
5.526E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-24	WCCI	nSPY
5.526E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-24	WCCI	SPRY-L
5.526E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-24	WCCI	SPRY-L
5.200E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	nCCI	nSPY
5.200E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	WCCI	SPRY-L
5.200E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	nCCI	SPRY-L
5.200E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-44	WCCI	nSPY
4.973E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-24	WCCI	nSPY
4.633E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	WCCI	SPRY-L
4.633E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	WCCI	nSPY
4.633E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	WCCI	SPRY-L
4.170E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	WCCI	nSPY
4.056E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-24	WCCI	nSPY
4.056E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-F-B	SPB-ny	RBB-24	WCCI	SPRY-L
3.890E-04	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	nCCI	SPRY-L
3.890E-04	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	nCCI	SPRY-L
3.890E-04	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	nCCI	nSPY
3.790E-04	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	DCCI	nSPY
3.614E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-24	WCCI	SPRY-L
3.614E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-24	WCCI	SPRY-L
3.614E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-24	WCCI	nSPY
3.501E-04	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	nCCI	nSPY
3.403E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-44	DCCI	nSPY
3.252E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-24	WCCI	nSPY
2.956E-04	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-24	DCCI	nSPY
2.898E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-44	DCCI	nSPY
2.791E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-22	nCCI	SPRY-L
2.791E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-22	nCCI	nSPY
2.791E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-22	nCCI	SPRY-L
2.651E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-ny	RBB-44	nCCI	nSPY
2.651E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-nB	SPB-ny	RBB-44	nCCI	SPRY-L
2.51[illegible]E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-22	nCCI	nSPY
2.487E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F	SPB-nn	RBB-24	DCCI	nSPY
2.261E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-24	DCCI	nSPY
1.981E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S	SPB-nn	RBB-22	WCCI	SPRY-L
1.981E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-22	WCCI	nSPY
1.981E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-22	WCCI	SPRY-L
1.913E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	DCCI	nSPY
1.739E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-44	DCCI	nSPY
1.587E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S	SPB-nn	RBB-22	WCCI	nSPY

Table A-8
Central Estimate Detailed Results For The TCSRV23 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
1.587E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-22	DCCI	nSPY
1.492E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	DCCI	nSPY
1.356E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-F-B	SPB-ny	RBB-24	DCCI	nSPY
1.356E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.356E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	nCCI	SPRY-L
1.356E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	nCCI	nSPY
1.221E-04	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	nCCI	nSPY
1.058E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	WCCI	nSPY
1.058E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-22	WCCI	SPRY-L
1.058E-04	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	WCCI	SPRY-L
9.797E-05	TC-V	CFD-E3P	CLC-WWD	CLV-333	CD-nILM	VBD-F	SPB-ny	RBB-22	DCCI	nSPY
9.521E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	DCCI	nSPY
9.521E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-22	WCCI	nSPY
6.376E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-44	nCCI	SPRY-L
6.376E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-44	WCCI	nSPY
6.376E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-44	WCCI	SPRY-L
6.376E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S	SPB-nn	RBB-44	WCCI	SPRY-L
6.376E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-44	nCCI	SPRY-L
6.376E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S	SPB-ny	RBB-44	nCCI	nSPY
5.738E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-44	DCCI	nSPY
5.738E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-44	nCCI	nSPY
5.738E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S	SPB-nn	RBB-44	WCCI	nSPY
5.171E-05	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-SE-B	SPB-ny	RBB-22	nCCI	SPRY-L
5.171E-05	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-SE-B	SPB-ny	RBB-22	nCCI	SPRY-L
5.171E-05	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-SE-B	SPB-ny	RBB-22	nCCI	nSPY
4.973E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-24	WCCI	SPRY-L
4.973E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S	SPB-nn	RBB-24	WCCI	SPRY-L
4.973E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-24	WCCI	nSPY
4.654E-05	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-SE-B	SPB-ny	RBB-22	nCCI	nSPY
4.476E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S	SPB-ny	RBB-24	DCCI	nSPY
4.476E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S	SPB-nn	RBB-24	WCCI	nSPY
4.034E-05	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-SE-B	SPB-ny	RBB-22	WCCI	SPRY-L
4.034E-05	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-ILM	VBD-SE-B	SPB-ny	RBB-22	WCCI	nSPY
4.034E-05	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-SE-B	SPB-ny	RBB-22	WCCI	SPRY-L
3.826E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	nCCI	nSPY
3.826E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	WCCI	nSPY
3.826E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	WCCI	SPRY-L
3.826E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	nCCI	SPRY-L
3.826E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	[illegible]	SPRY-L
3.826E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-44	[illegible]	SPRY-L
3.630E-05	TC-V	CFD-E3P	CLC-DDD	CLV-333	CD-nILM	VBD-SE-B	SPB-ny	RBB-22	[illegible]	nSPY
3.443E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	DCCI	nSPY
3.443E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	WCCI	nSPY
3.443E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-44	nCCI	nSPY
2.984E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-24	WCCI	nSPY

11L

Table A-8
Central Estimate Detailed Results For The TCSRV23 Plant Damage State

Conditional Probability	Scenario	Containment Failure Mode	Containment Failure Location	Containment Failure Level	In-Vessel Progression	Vessel Breach Mode	Suppression Pool Bypass	Reactor Building Bypass	Core-Concrete Interation	Status of Drywell Sprays
2.984E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S-B	SPB-ny	RBB-24	WCCI	SPRY-L
2.984E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-24	WCCI	SPRY-L
2.686E-05	TC-V	CFD-E3P	CLC-WDD	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-24	DCCI	nSPY
2.686E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S-B	SPB-ny	RBB-24	WCCI	nSPY
2.512E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-22	nCCI	nSPY
2.512E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-ILM	VBD-S	SPB-nn	RBB-22	nCCI	SPRY-L
2.512E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S	SPB-nn	RBB-22	nCCI	SPRY-L
2.261E-05	TC-V	CFD-E3P	CLC-WWW	CLV-333	CD-nILM	VBD-S	SPB-nn	RBB-22	nCCI	nSPY

9.9498E-01 TOTAL FREQUENCY OF BINS PRINTED

TOTAL FREQUENCY IN ALL BINS = 1.0000E+00
TOTAL FREQUENCY LOST DUE TO TERMINATION OF BRANCHES 5.0154E-03
TOTAL FREQUENCY FOR THIS SEQUENCE/WALKTHROUGH = 1.0050E+00

115

APPENDIX B

Detailed Description of the Peach Bottom Containment Event Tree

C. N. Amos
(Technadyne Engineering Consultants, Inc.)

J. M. Griesmeyer
(Sandia National Laboratories)

May 5, 1987

INTRODUCTION

This appendix describes the computerized containment event tree (CBT) for the Peach Bottom Atomic Power Station (PBAPS), Unit 2. This event tree was developed at Sandia National Laboratories in conjunction with the Severe Accident Risk Reduction Program (SARRP). Previous reports on the CET for Surry [1] and Sequoyah [2] provide additional information on the development of the CET analysis.

Peach Bottom is a boiling water reactor (BWR) designed by the General Electric Company. The nuclear steam supply system (NSSS) is of the BWR/4 product line. PBAPS has a Mark I containment design which incorporates a pressure suppression design with a relatively small free volume in containment (approximately one-tenth the containment volume of a pressurized water reactor with a large-dry containment). The BWR/4 does have a large number of systems which are capable of injecting water into the reactor core. There are two systems that provide high-pressure make-up: Reactor Core Isolation Cooling (RCIC) and High-Pressure Coolant Injection (HPCI). These are both steam-turbine-driven and are independent of ac power. In addition, long-term (after 8 to 10 hours) high-pressure make-up can be accomplished by flow from the Control Rod Drive (CRD) hydraulic system pumps. There are four low-pressure Emergency Core Cooling System (ECCS) pumps: two Low-Pressure Core Spray (LPCS) pumps, which deliver water directly on top of the core, and two Low-Pressure Coolant Injection (LPCI) pumps. Feedwater is supplied by the condensate and feedwater pumps. The feedwater pumps are steam-turbine-driven (at Peach Bottom) and are not available in many accident scenarios. The condensate pumps could be used for low-pressure injection. Finally, Peach Bottom has a cross-tie between the High-Pressure Service Water (HPSW) system and LPCI. Thus, HPSW is yet another means of low-pressure injection. The CET considers all of the systems.

The EVNTRE computer program (developed by J. M. Griesmeyer and V. L. Behr of Sandia National Laboratories) allows the evaluation of event trees that have sufficient complexity to adequately model accident progression and include provisions for representing uncertainty in data, modeling, and phenomena. Thus, a containment event tree has been developed for PBAPS that adequately represents all credible progression scenarios for transient-initiated severe accidents. The tree was not developed to consider LOCAs (although some limited provision has been made for their analysis) since the frequency of core damage from LOCAs was determined by the Accident Sequence Evaluation Program (ASEP) to be sufficiently low that they represent a negligible contribution to offsite risk. The EVNTRE code is essentially a bookkeeping program that keeps track of the numerous possible pathways through the tree and, given the probability of each possible branch, calculates the probability associated with each possible pathway. For a tree which is as complex as those developed for SARRP, there are literally thousands of possible pathways through the tree (each pathway represents a possible progression scenario for the accident). EVNTRE thus includes the capability to "bin" these outcomes, i.e., to group them according to certain key parameters selected by the user. For this application the parameters selected are those which are relevant to the accident source term. In previous PRAs, binning could be as simple as early, intermediate, and late containment failure. EVNTRE allows many more binning dimensions or characteristics. For example, the binning dimensions used in the current analysis include: the size of leakage for early containment failure, the size of leakage for late containment failure, and the extent to which the suppression pool is bypassed. The number of dimensions allowed and the flexibility in defining them allow a careful evaluation of uncertainties and sensitivities and also provide a tool that is extremely powerful for evaluating the impact of proposed methods for reducing risk.

The PBAPS CET is composed of 107 questions. The majority of these represent branch points in the tree that may have several alternate branches (up to five in this tree). In this respect, the CET differs from typical event trees that are restricted to only two outcomes for each branch point. In fact, most of the questions have only two branches. The questions that do not represent branch points are used to introduce quantitative information (via "parameters") into the tree. An example of such quantitative information would be the expected pressure rise in containment due to hydrogen produced in-vessel. The probability of a given branch can be dependent on the branches taken in previous questions. Thus, the probability that the drywell shell is melted by core debris at vessel breach can be dependent on whether or not there is any water on the drywell floor. To determine the appropriate set of probabilities, the path through previous branches in the tree is compared against Boolean expressions involving the occurrence or absence of particular branches at previous questions on the tree. The probabilities for the first case encountered that produces a match are used. This point is extremely important in interpreting the containment event tree presented in this appendix. For most questions the CET (which is actually the input for the EVNTRE code) specifies the case-dependent branch point probabilities. The code does, however, have the capability to calculate branch point probabilities internally. This capability is utilized, for example, in the calculation of the containment failure probability. The code carries the values of "parameters," which are specified in a case-dependent manner through the CET. Parameters such as the pressure in containment at the onset of core damage, and the pressure rise due to hydrogen production, are summed and compared to a user input fragility curve (which could also be case dependent reflecting temperature effects, for example). This comparison determines the branch point probability for the question.

The code proceeds through the CET and evaluates the probability of each possible pathway, where the pathway refers to the complete combination of events. Typically, the user specifies a cut-off probability such that if the probability of a pathway drops below the cut-off after a given branch, then further exploration of the pathway is abandoned. The cut-off probability (conditional on core melt) is specified at $1x10^{-5}$ in the current analysis. This cut-off probability is very low and should not be interpreted to mean that the analysts believe that containment performance is currently understood at that level of detail. This cut-off is determined such that the number of pathways that are "lost" (i.e., do not turn up in the binned results) is held to a minimum. Thus, the conditional probabilit3 of all the accident progressions resulting from a given plant damage state is close to unity. The pathways with a conditional probability greater than the cut-off are binned according to the features of the pathway that are relevant to evaluating the source term (the binning dimensions). A subset of the binning dimensions used is presented in Section 2 of this report.

For SARRP, the quantification of the CET pathways has been performed for three different evaluations of the branch point probabilities for each question and case. The branch point probabilities have been evaluated based on an optimistic, a

118

central, and a pessimistic estimate (termed the OCP approach). The central estimate reflects a combination of the authors' best estimates of the probabilities, based on an understanding of the plant and the phenomena involved, together with a composite of current thinking among researchers in the light-water reactor (LWR) severe accident area. The central quantification draws heavily on the opinions of experts in various areas of LWR severe accident phenomenology. The optimistic evaluation gives what the authors judge to be the greatest reasonable probabilities to branches that are expected to result in the lowest offsite consequences. The pessimistic evaluation gives what the author judges to be the greatest reasonable probability to the branch of a given question that is expected to result in the greatest offsite consequences. The tree is quantified three times: once assuming all optimistic evaluations (subject to consistency constraints), once assuming all central evaluations, and once with all pessimistic evaluations. Each quantification has been termed a "walkthrough." This approach is intended to give a feel for the range of possible results. However, it is recognized that there are some deficiencies that limit its usefulness. Due to time constraints, more effort has been placed into making the central evaluations than the optimistic and pessimistic evaluations. Thus, documentation of the optimistic and pessimistic evaluations in this appendix is also minimal.

Uncertainty in the results of the CET analysis (as well as other portions of the risk evaluation for Peach Bottom) has been studied employing a limited application of Latin Hypercube sampling (LHS). This Monte-Carlo type approach to uncertainty analysis made use of the EVNTRE code sampling capability. Sampling involves simultaneously varying sets of individual (case dependent) branch probabilities, while leaving the structure of the tree untouched. For each sample observation, the values for the probabilities are randomly selected from within their range (but allowing only certain discrete values). The resulting distribution of the outputs (accident progression bins) provides information regarding uncertainties, and exploration of the relationships between the inputs and outputs provides sensitivity information.

This document provides a detailed description of the PBAPS CET. The description begins with a summary of the major sections of the tree. A question-by-question and case-by-case description of the structure and logic follows. The EVNTRE input (the actual CET) is provided in Appendix E. Appendix D demonstrates, by example, how to interpret the EVNTRE input.

The rationale used in the quantification of the branch point probabilities and code parameters is discussed in this appendix (at least for the central estimate). The CET considers many phenomenological issues that are not well understood at the current time. Quantifying the probability that such phenomena will (or will not) occur is thus a subjective process. The general rule of thumb that has been used is as follows: If a particular phenomenon is completely uncertain, that is, there is no reason to believe one outcome more than another in the authors' opinion, the outcome was assigned a probability of 0.5 in the central walkthrough (these issues are always considered in a binary, yes-or-no context). A probability of 0.9 was assigned in the optimistic walkthrough to that outcome, which would tend (in the author's judgment) to reduce offsite consequences, and a probability of 0.9 was assigned in the pessimistic walkthrough to the opposite outcome. If, on the other hand, an issue is uncertain but it was the author's judgment that there was some reason to believe that a particular outcome was more likely, this outcome was given a 0.9 in the central walkthrough, and the 0.5 probability was assigned to the optimistic or pessimistic walkthrough, as appropriate. In the remaining walkthrough, a probability between 0.99 and 1.0 was assigned depending upon the author's judgment as to the weight of the evidence suggesting that the particular outcome was more likely.

Questions #1 through #16 define the initial and boundary conditions for the accident progression analysis (i.e., the plant damage state). The branches taken for these questions are determined by inspection of the cut sets developed under ASEP for the accident sequence under consideration. The ASEP cut sets for the various accident sequences are grouped according to the initial and boundary conditions that they imply for evaluation of the CET. These questions thus form the interface between the front-end analysis and the CET.

The remaining questions on the tree (#17 through #107) have branch point probability evaluations that are dependent on the branches taken in previous questions and that normally do not change as a function of the cut set being considered. The exception to this statement lies in the use of the CET to resolve sequences which were determined by ASEP to be core-vulnerable. Core-vulnerable sequences result in a challenge to containment integrity, but not necessarily to core-cooling. The CET was used to quantify what fraction, if any, of these core-vulnerable sequences resulted in core damage. For the purpose of that exercise, branch point probabilities were adjusted for some of the damage states. Those values are not preserved in the current tree.

The event tree is time-phased. Several questions are repeated in the tree at various times in the accident progression. The availability of pumps to supply water to the reactor vessel or containment is considered at five different times in the tree. Of course, each time regime has its own unique questions. For instance, it only makes sense to ask what the pressure rise in containment due to Reactor Pressure Vessel (RPV) blowdown was immediately after RPV failure. The questions that are asked are divided into groups that cover distinct time regimes in the accident. These time regimes are (1) the initial conditions which can lead to core uncovery (an E1 precedes the branch names); (2) the period while the operators are attempting to avert core damage (generally an E2 is used in the branch name); (3) the period immediately before core damage, which establishes the conditions under which core damage occurs (an E3 is used in the branch name); (4) the period during core damage (generally an E4 precedes the branch name); (5) the time immediately preceding RPV meltthrough or in-vessel recovery (E5 designator); (6) the events associated with vessel breach (the letter I appears frequently at the start of the branch name); and (7) the period following vessel breach (termed late, an L precedes many of the branch names). Questions #1 through #16 form the "E1" period. Questions #17 through #21 complete the sequence definition ("E2" period) by considering events and actions which are implied by the ASEP results but not specifically quantified. Questions #22 through #31 ("E3" period) consider system availability and the integrity of the plant so as to determine whether the accident can be recovered prior to core melt. Questions #31 through #34 form a transition between consideration of conditions in the plant prior to core damage and consideration of conditions at the onset of core damage. Recovery prior to core melt is considered in Question #35 which in a sense determines the separation between the "E3" and "E4" time periods. Questions #36 through #53 determine the accident progression during core degradation (the "E4" period). System availability and plant integrity at the time of vessel breach are established by Questions #54 through #67. Question #69 allows for accident recovery prior to vessel breach and marks the end of the "E5" time period. The next 14 questions determine accident progression at vessel breach (Questions #70 through #83), which is the "I" time period. The late time period ("L"), after vessel breach, is considered by Questions #84 through #107.

Definition of Question Types

The EVNTRE Code considers six types of questions (branch points). These are described as follows:

Type 1	Branch point probabilities are input and are independent of the branches taken in previous questions.
Type 2	Branch point probabilities are input and are dependent on branches taken in previous questions.
Type 3	Used to input parameter values to the code. Branch point probabilities are also input.

121

Type 4 Used to input parameter values to the code. Branch point probabilities are also input. They are dependent on the branches taken in previous questions, as are the parameter values.

Type 5 Branch point probabilities are calculated by the code. The input is a list of parameters to be summed and values of reference parameters to which the sum is compared.

Type 6 Branch point probabilities are calculated by the code. The input is a list of parameters to be summed and values of reference parameters to which the sum is compared. The list, the reference parameters, and the method of comparison are dependent on the branches taken in previous questions.

Probabilities

SARRP has adopted descriptive terms for the probabilities of occurrence that are assigned to various events and phenomena. Since many of these probability assignments are subjective, the verbal descriptions are generally more appropriate than the statement of numerical value. The range of numerical values associated with each descriptor and the representative value used are given below:

Certain: Probability = 1.0

Highly Likely: 1.0 > Probability > 0.995 (Used 0.999)

Very Likely: 0.995 > Probability > 0.95 (Used 0.99)

Likely: 0.95 > Probability > 0.7 (Used 0.9)

Indeterminate: 0.7 > Probability > 0.3 (Used 0.5)

Unlikely: 0.3 > Probability > 0.05 (Used 0.1)

Very Unlikely: 0.05 > Probability > 0.005 (Used 0.01)

Highly Unlikely: 0.005 > Probability > 0.0 (Used 0.001)

Impossible: Probability = 0.0 (Negligible)

A detailed description of the event tree structure (questions and cases) follows. The quantification for Questions #1 through #16 corresponds to plant damage state (cut set group) TBUX.

QUESTION #1: What is the initiating event?

Defines the initiating event. These events are those initiators identified by ASEP as being the most probable contributors to overall risk (i.e., frequency > 1E-7). The branch point probabilities will be zero or unity (for only one branch, depending upon which plant damage state is to be analyzed by the tree).

This is a Type 1 question, and there are seven branches:

1. A: Initiator is a large-break LOCA.

2. S1: Initiator is a small break that is in the size range such that High Pressure Coolant Injection (HPCI) cannot maintain level and therefore the Automatic Depressurization System (ADS) must be used to allow low-pressure coolant injection.

3. S2/3: Initiator is a small break that is small enough that HPCI provides adequate injection to prevent loss of RPV level to the point that ADS is actuated, (i.e., level is maintained above Level 1) (ASEP frequency < 1 E-7).

4. TLOSP: Initiator is a transient that is either induced by or results in the loss of offsite power.

6. TPCS: Initiator is a transient induced by turbine trip or closure of the Main Steam Isolation Valve(s) (MSIV). The Power Conversion System is assumed to be isolated from the Reactor Pressure Vessel (RPV).

6. TC: Initiator is some transient followed by a failure of the Reactor Protection System (RPS) such that no scram occurs.

7. IORV: The initiator is the Inadvertent Opening of a Relief Valve by the operator.

QUANTIFICATION:

For this plant damage state:

TLOSP: 100% of the cut sets

QUESTION #2: What is the initial break location?

Specifies the pathway of effluent steam (or hydrogen/fission products) from the RPV to the containment. This question is not referenced in the current tree. For some plants and for certain cut sets, this question could be used in determining success criteria. It is included here for the sake of completeness.

This is a Type 1 question, and there are five branches:

1. RPV: Break is in piping connected to the RPV but not in the Recirculation Lines, Main Steam Lines (MSL) of Feedwater (FW) lines.
2. Rcirc: Break is in the Recirculation line
3. MSL: Break is in the Main Steam Line.
4. FW: Break is in the Feedwater line.
5. SRV: There is no break. Steam leaves the RPV only through the Safety/Relief Valves.

QUANTIFICATION:

For this plant damage state:

SRV: 100% of the cut sets

QUESTION #3: What is the level of preexisting containment leakage or isolation failure?

BWR operating history indicates some instances where plants were operating with leakage to the environment or the reactor building in excess of the technical specification limits. This excessive leakage was usually a result of a failure to properly close primary containment penetrations. It is believed, and recent data substantiates, that this is not a concern with inerted containments. The Peach Bottom containment carries an overpressure of nitrogen with respect to ambient pressure that would result in rapid detection of any large leakage. Leakage in excess of technical specifications is thus deemed impossible. This question is not referenced in the current version of the tree. Thus, branch 6 (no excessive leakage) is always taken (probability of 1.000).

This is a Type 1 question, and there are five branches:

1. E1L2D: There is preexisting leakage from the drywell large enough to remove decay heat from the containment 10 minutes after shutdown.

2. E1L1D: There is preexisting leakage from the drywell in excess of technical specification limits but not sufficiently large to remove all the decay heat generated.

3. E1L2W: There is preexisting leakage from the wetwell large enough to remove decay heat from the containment 10 minutes after shutdown.

4. E1L1W: There is preexisting leakage from the wetwell in excess of technical specification limits but not sufficiently large to remove all the decay heat generated.

6. E1LO: There is no leakage from the containment in excess of technical specifications (<0.5% of the containment free volume per day).

QUANTIFICATION:

For this plant damage state:

E1LO: 100% of the cut sets

QUESTION #4: Is there a loss of offsite power?

Determines if there is power available to ac-powered equipment that is not connected to the emergency busses. Probability of LOSP is unity if the initiating event is TLOSP.

This is a Type 1 question, and there are two branches:

1. E1-LOP: There is a loss of offsite power.

2. E1nLOP: Offsite power is available.

QUANTIFICATION:

For this plant damage state:

E1-LOP: 100% of the cut sets

QUESTION #5: Is there a Station Blackout (diesel generators fail)?

Determines if there is power on the emergency busses. A station blackout means that there is no offsite power and the diesel-generators are not supplying power to the emergency busses (i.e., neither of the two diesel generators is running). The branch taken is determined from the plant damage state to be analyzed.

This is a Type 1 question, and there are two branches:

1. SB: There is a station blackout. No Ac power is available even on the emergency buses.

2. nSB: There is not a station blackout. Ac power is available at least from the emergency buses.

QUANTIFICATION:

For this plant damage state:

SB: 100% of the cut sets

QUESTION #6: Is dc power not available?

This question determines the availability of the station batteries. The station battery supplies ac power to the following critical equipment: (1) Vessel instrumentation (level, pressure, etc.), (2) Pilot valves on the Safety/Relief Valves (SRVs) (allow automatic depressurization and relief function), (3) Valves, instrumentation and control for vessel injection from the High-Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, (4) Diesel-Generator starters, and (5) Switches and circuit breakers (required for the recovery of power to the station buses). ASEP determined common-mode failure of the station batteries as being a significant contributor to the frequency of station blackout events. This question is thus highly significant in the determination of the plant damage state.

This is a Type 1 question, and there are two branches:

1. ElfDC: The station batteries are failed. There is no dc power.

2. El-DC: Power is available on the dc buses.

QUANTIFICATION:

For this plant damage state:

ElfDC: 100% of the cut sets

QUESTION #7: For TC, does SLC fail to inject?

This question applies only to sequences involving failure of the Reactor Protection System (RPS) scram or Alternate Rod Insertion (ARI) functions (i.e., the initiator is an ATWS). In such sequences, the Standby Liquid Control (SLC) system provides a means to achieve plant shutdown. SLC is manually actuated and consists of two pumps (only one at a time may operate) that inject enriched sodium pentaborate solution into the RPV. The solution is enriched in boron-10 (relative to natural boron) so as to achieve an 86 gallons-per-minute equivalent flow. The branch taken is determined by ASEP cut sets. It must be noted that these cut sets are for SLC operation within 4 minutes of the initiating transient. The assumption that 4 minutes is acceptable is based on calculations using the BWR-LTAS computer code. If injection were delayed beyond this time, core damage might still be prevented but additional actions on the part of the operator would be required. One may consider such delayed recovery by the operating staff as being accounted for in the containment event tree. Some fraction of the sequences identified by ASEP as being core-vulnerable are shown not to result in core damage (eventual SLC actuation, manual rod insertion, or scram recovery is assumed but is not explicit in this tree).

This is a Type 1 question, and there are three branches:

1. ElfSLC: SLC is not initiated within 4 minutes of the transient.

2. El-SLC: SLC is initiated within 4 minutes of the transient.

3. na: The event is not a TC, the RPS performed correctly.

QUANTIFICATION:

For this plant damage state:

na: 100% of the cut sets

QUESTION #8: Does a SRV stick open early?

This question determines whether there is a failure of a least one Safety/Relief Valve (SRV) to reseat after operation (P failure). Branch point probabilities are determined by ASEP and are assigned in accordance with the plant damage state being analyzed.

This is a Type 1 question, and there are two branches:

1. El-SORV: There is at least one stuck-open relief valve.

2. ElnSORV: There are no stuck-open relief valves.

QUANTIFICATION

For this plant damage state:

ElnSORV: 100% of the cut sets

QUESTION #9: Do the HPCI and RCIC systems fail to inject?

Determines whether the High-Pressure Coolant Injection (HPCI) and/or Reactor Core Isolation Cooling (RCIC) systems are operating initially. Both systems are driven by steam turbines that draw steam from the Main Steam Lines. Turbine exhaust is dumped to the suppression pool. Suction for both systems is initially from the Condensate Storage Tank (CST). Suction will switch to the suppression pool on a low-CST-level signal. Both systems require dc power to function. Both systems will isolate and stop injecting on high containment pressure (>40 psig for RCIC), the pressure at which HPCI isolates is greater than the assumed failure pressure of 132 psia). HPCI also isolates on low RPV pressure (<150 psia). ASEP assumed that HPCI would fail when the water being pumped from the suppression pool exceeded 200 to 250°F; this failure mode is significant to the majority of plant damage states initiated by ATWS. The branch taken is dependent on the plant damage state being considered.

This is a Type 1 question, and there are three branches:

1. ElfHPC: Both high-pressure cooling systems (HPCI and RCIC) have failed.
2. ElrHPC: Neither HPCI or RCIC is currently providing water injection to the RPV. However, at least one of these systems can be recovered.
3. El-HPC: Either HPCI or RCIC (or most likely both) is providing high-pressure make-up water to the vessel.

QUANTIFICATION:

For this plant damage state:

ElfHPC: 100% of the cut sets

QUESTION #10: Does the CRD hydraulic system inject?

Determines the availability of the Control Rod Drive (CRD) hydraulic system for injecting water into the vessel. The

128

CRD system provides cooling flow to the control rod drives during normal operation. Normal flow is approximately 56 gpm with the reactor at hot standby (zero power but at pressure) normal flow is 110 gpm. Suction for CRD flow is drawn from the CST. The CRD pump head can be up to 1,800 psi. Using both CRD pumps, plant operators can manipulate the system to attain flows up to 200 gpm; with the RPV depressurized, over 300 gpm is possible. The branch taken is determined by the ASEP cut sets of interest. It should be noted that the CRD pumps are normally powered off the station bus (offsite ac). At Peach Bottom, emergency power (diesels) may be used.

This is a Type 1 question, and there are three branches:

1. ElfCRD: The CRD cooling system is failed.

2. ElrCRD: The CRD cooling system is not currently operating but is recoverable. This situation could arise in a loss of offsite power.

3. El-CRD: The CRD system is providing high-pressure make-up.

QUANTIFICATION:

For this plant damage state:

ElrCRD: 100% of the cut sets

QUESTION #11: Do the LPCS and LPCI systems fail?

Determines whether low-pressure cooling systems are available to inject water to the RPV. There are two low-pressure Emergency Core Cooling (ECC) systems: (1) the Low Pressure Core Spray (LPCS) and Low Pressure Coolant Injection (LPCI). The two the LPCI pumps are also the Residual Heat Removal (RHR) system pumps and thus provide shutdown cooling, suppression pool cooling, and containment (drywell) spray. (The LPCI pumps also supply torus spray, but that function is not of interest here.) The low-pressure ECC system (LPCS and LPCI) for each of the two units at Peach Bottom consists of four pumps divided such that two pumps from each system are powered by a separate electrical division. Thus, there are two low-pressure pumps powered by each diesel. Each LPCS pump has a rated flow of 3125 gpm; run-out flow is in excess of 5,000 gpm. The LPCI pumps have a nominal flow of 10,000 gpm. Suction for these pumps is normally drawn from the suppression pool but can be taken from the CST in the case of LPCS. High-pressure interlocks prevent the injection valves from opening with the RPV above 450 psi. Branch point probabilities are determined by the plant damage state being considered. It is noted that none of the damage states results from failure of all low-pressure ECCS.

This is a Type 1 question, and there are four branches:

1. ElfLPC: Both LPCS & LPCI are failed.

2. ElrLPC: Both LPCS & LPCI are not available for injection to the vessel. However, at least one of these systems is recoverable. This situation could arise in a station blackout.

3. ElaLPC: Either LPCS or LPCI (or both) is available (i.e., the pumps are running), but there is no injection to the vessel (injection valves are closed either due to high RPV pressure or system failure).

4. El-LPC: Either LPCS or LPCI (or both) is providing water injection to the vessel.

QUANTIFICATION:

For this plant damage state:

ElrLPC: 100% of the cut sets

QUESTION #12: Do the RHR systems fail?

Suppression Pool Cooling (SPC) and the Containment Spray System (CSS) are two modes of the Residual Heat Removal (RHR) system. LPCI (see above) is also a mode of RHR. In either the SPC or CSS modes of operation, the RHR system can remove heat from the suppression pool by passing water from the pool through heat exchangers (with service water on the shell side). In spray mode, water is sprayed into the drywell. Spray water can return to the suppression pool once the water level in the drywell reaches the bottom of the vent pipes (approximately 18 inches deep). Both SPC and CSS must be manually initiated. Following a LOCA signal (low RPV level and high drywell pressure), automatic interlocks prevent actuation of SPC or CSS for 10 minutes so as to assure LPCI availability to provide water to the RPV. The branch point probabilities are determined by the plant damage states that are being considered.

This is a Type 1 question, and there are three branches:

1. ElfRHR: Both SPC and CSS are failed.

2. ElrRHR: Both SPC and CSS are not currently available but are recoverable. This condition could occur in a station blackout.

3. El-RHR: Either SPC or CSS (or most likely both) are available.

QUANTIFICATION:

For this plant damage state:

ElrRHR: 100% of the cut sets

QUESTION #13: Does the condensate system fail?

Determines the availability of the condensate system. Since the feedwater pumps are turbine driven, closure of the Main Steam Isolation Valves (MSIVs) results in the loss of feedwater (i.e., the ability to inject condensate at high pressure). However, condensate pumps are motor driven and thus the condensate system can be available whenever off-site power is available. Injection is through the feedwater lines to the feedwater sparger inside the vessel. The condensate system can thus deliver water when vessel pressures are up to 600 psi. Since this is a high-pressure system, there are no pressure interlocks, and isolation is done manually if at all. Suction is taken from the condenser hotwell. The branch point probabilities are determined by the plant damage state that is being analyzed.

This is a Type 1 question. and there are three branches:

1. ElfCOND: The condensate system is failed.

2. ElrCOND: The condensate system is unavailable but can be recovered. Such a condition would result from a loss of offsite power (for example).

3. ElaCOND: The condensate system is available for injection to the RPV but is not necessarily in use.

QUANTIFICATION:

For this plant damage state:

ElrCOND: 100% of the cut sets

QUESTION #14: Does HPSW fail in a mode that would preclude injection?

Determines the availability of the High-Pressure Service Water system for providing injection to the vessel through a cross-tie with the RHR (shutdown cooling mode) system. HPSW normally provides cooling flow for the RHR heat exchangers. The system has four pumps, two powered by each emergency electrical division. Suction is taken essentially from the river. The branch point probabilities are determined by the ASEP cut sets of interest.

This is a Type 1 question, and there are three branches:

1. ElfHPSW: HPSW is not available for injection to the vessel.
2. ElrHPSW: HPSW is unavailable but may be recovered. Loss of offsite power is the most likely cause of this condition.
3. ElaHPSW: HPSW is available. Operator action is required to align this system for injection to the vessel.

QUANTIFICATION:

For this plant damage state:

ElrHPSW: 100% of the cut sets

QUESTION #15: Is there a failure that precludes drywell cooler operation?

Determines the availability of drywell coolers for containment heat removal. The coolers are sized to remove the drywell heat load during normal operation. This load is approximately 1 megawatt. While the drywell coolers are inadequate to remove decay heat, their operation could impact accident progression. The branch taken is determined by the plant damage state being analyzed. This question is not referenced in the current tree.

This is a Type 1 question, and there are three branches:

1. ElfDWC: The drywell coolers are failed.
2. ElrDWC: The drywell coolers are unavailable but may be recovered.
3. ElaDWC: The drywell coolers are available.

QUANTIFICATION:

For this plant damage state:

ElrDWC: 100% of the cut sets

QUESTION #16: Is ADS blocked or not called upon?

The Automatic Depressurization System (ADS) is designed to lower the RPV pressure in the event that the high-pressure systems are not maintaining adequate vessel level. This situation may occur during a small-break LOCA. ADS logic requires three indications to initiate: (1) low RPV level (Level 1), (2) at least one LPCS or LPCI pump running (discharge pressure indication), and (3) the ADS timer (actuated on Level 1 signal) is timed out. The Peach Bottom ADS timer runs for 8 minutes. In an ATWS event the Emergency Operating Procedures (EOPs) dictate that ADS should be inhibited. The reason an 8-minute timer is incorporated is to allow time for the operators to inhibit the ADS. Thus, no ADS is interpreted as success for ATWS. ADS may also be actuated manually. However, manual depressurization is more likely to occur through manual actuation of the SRVs in relief mode. ADS relief valves (five) have their own air accumulators (which at Peach Bottom can be charged using nitrogen cylinders that are independent of the normal plant nitrogen system) and require only dc power to the solenoid valve on the accumulator to hold the associated valve open. The branch point probabilities are determined by the plant damage state that is to be analyzed. For this question, manual depressurization under circumstances that would have led to ADS will be taken to be the same as ADS.

This is a Type 1 question, and there are two branches:

1. ElnADS: There has been no automatic depressurization of the vessel.
2. El-ADS: Automatic (or equivalent manual) depressurization

QUANTIFICATION:

For this plant damage state:

ElnADS: 100% of the cut sets

QUESTION #17: Does the RPV remain pressurized?

Determines the RPV pressure once the plant damage state has been established. The question accounts for primary system breaks as well as depressurization as described in Question #16. Emergency Operating Procedures (EOPs) prescribe vessel depressurization in circumstances when ADS would not occur, in particular, when the suppression pool temperature rises above the predetermined limit with respect to RPV pressure. This limit is the High-Temperature Condensation Limit (HTCL), which is designed to avoid condensation oscillation dynamic loads in the pool that are postulated to occur when steam is condensed in a hot suppression pool. During isolation events when no ECC injection (or RCIC) is available, depressurization is included in the procedure to prolong core cooling and provide for the use of the condensate system or HPSW for injection to the vessel.

This is a Type 2 question, and there are two branches:

1. E2nDeP: The vessel is not depressurized prior to core damage (if any).
2. E2-DeP: The vessel is depressurized prior to core damage (if any).

CASE #1:

In this case, the RPV has been depressurized either through a large break or through the use of ADS. The RPV is thus certainly depressurized.

QUANTIFICATION:

	Optimistic	Central	Pessimistic
E2nDeP:	Impossible	Impossible	Impossible
E2-DeP:	Certain	Certain	Certain

CASE #2:

In this case, at least one SRV is stuck open. For the dominant cut sets SPC is failed and low pressure systems are available. The suppression pool is heating up. The open SRV and the action of HPCI will be lowering the vessel pressure. The Emergency Operating Procedures (EOPs) instruct the operator to depressurize the vessel to stay within the HTCL limit (described above). In addition, the operators are trained to depressurize the RPV in order to attempt reseating of the stuck open valve. Consideration is given to both these motivations in assigning the branch point probabilities for this question. The quantification is based on the judgment of the analyst.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2nDeP:	Impossible	Highly Unlikely	Unlikely
E2-DeP:	Certain	Highly Likely	Likely

CASE #3:

In this case there is a station blackout with initial high-pressure injection (HPCI/RCIC). The emergency procedures would lead the operators to depressurize the vessel to maintain the vessel pressure below that which is required by the HTCL curve. Consideration is given to the fact that coolant injection comes only from turbine-driven systems. There may thus be concern over the loss of high-pressure systems should the vessel be depressurized. It is recognized that depressurization to 150 to 200 psi will not cause isolation of HPCI or RCIC. Branch point probability assignments are consistent with values used in the AEP portion of this work for manual depressurization.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2nDeP:	Impossible	Highly Unlikely	Unlikely
E2-DeP:	Certain	Highly Likely	Likely

CASE #4:

In this case, there is a station blackout and dc power has failed. There is thus no power to operate the ADS system or use the relief mode of the SRVs. The RPV must remain at the safety valve setpoint (close to operating pressure).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2nDeP:	Certain	Certain	Certain
E2-DeP:	Impossible	Impossible	Impossible

CASE #5:

This case includes station blackout with failure of all high-pressure injection but with dc power available (all other blackout damage states were considered in Cases 3 and 4), and transient-initiated damage states with failure of all high-pressure injection. The two can probably be distinguished based on the availability of condensate or HPSW for injection. Such distinction is not made even though the operators may be more inclined to depressurize with one of these systems available. The EOPs dictate depressurization under either circumstance. In the blackout case, the motivation is to prolong core cooling for as long as possible. Since the procedures are symptom-based (and since the case in which condensate or HPSW is available is not part of the ASEP core damage profile), a single quantification is employed. This quantification is the opinion of the analyst but is consistent with values used by ASEP.

QUANTIFICATION:

	Optimistic	Central	Pessimistic
E2nDeP:	Impossible	Very Unlikely	Indeterminate
E2-DeP:	Certain	Very Likely	Indeterminate

CASE #6:

In this case there was a transient initiator and high-pressure injection is provided by HPCI, RCIC, or both. Failure of RHR is implied since, with containment heat removal available, such a sequence is not core-vulnerable. Since the two high-pressure make-up systems will eventually fail without containment heat removal, low-pressure systems provide the only means of eventual recovery. However, the operator should be following the emergency procedures and depressurize the vessel to remain within the HTCL curve. CRD flow is not available so the operators could not even consider relying on this as a means of high-pressure injection. The quantification of the branch points is based on the opinion of the analyst.

QUANTIFICATION:

	Optimistic	Central	Pessimistic
E2nDeP:	Impossible	Very Unlikely	Indeterminate
E2-DeP:	Certain	Very Likely	Indeterminate

CASE7:

This case is similar to the previous one except that CRD flow is available. This case includes the ASEP TW failure of containment heat removal sequences in which high-pressure injection is relied upon in the initial stages. CRD flow can provide adequate long-term cooling. Since the operator could choose to rely on CRD flow and ignore the HTCL limit, some slightly larger probability for the operator failing to depressurize might be assumed. However, the quantification here is essentially identical to the first (but for the optimistic evaluation).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2nDeP:	Highly Unlikely	Very Unlikely	Indeterminate
E2-DeP:	Highly Likely	Very Likely	Indeterminate

CASE #8:

This case includes all ATWS sequences (TC). ASEP has made an assessment of the probability of controlled depressurization in a TC sequence. The branch point probabilities for this case are thus either zero or unity, depending on the plant damage state that is being analyzed by the tree. For the majority of the TC sequences, the operators depressurize the RPV. However, the human error probabilities (HEPs), which were determined by the Human Reliability Analysis (HRA) carried out by Brookhaven National Laboratory (BNL) specifically for ATWS events at Peach Bottom, indicate a nonzero probability for the failure to depressurize prior to core uncovery resulting from the failure of the high-pressure systems (HPCI is the only system which can provide make-up for ATWS). Thus, all the ATWS-initiated plant damage states (except for TCSRX2 and TCSRV23) include ADS being inhibited and failure to depressurize when high-pressure injection is lost. The other ATWS sequences do not contribute significantly to core damage because it was judged likely that the operators would be successful in restoring (and maintaining) injection once the RPV was depressurized.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2nDeP:	Certain	Certain	Certain
E2-DeP:	Impossible	Impossible	Impossible

CASE #9:

This case covers transient initiated sequences with no high-pressure injection but with low pressure systems available. Depressurization is certain for these cases. Probably, the operators would manually depressurize (not waiting for the ADS timer). If there were no manual depressurization, ADS would occur. This case does not arise in any of the ASEP plant damage states.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2nDeP:	Impossible	Impossible	Impossible
E2-DeP:	Certain	Certain	Certain

QUESTION #18: For TC, is inadequate level maintained with LPI?

This question provides ASEP results for TC sequences to the tree. All six TC sequences defined by ASEP are considered. Branch probabilities are from ASEP results for the cut sets (or plant damage states) being considered. The question pertains only to the use of low-pressure systems for injection since, under the ASEP assumptions, the turbine-driven high-pressure systems (HPCI & RCIC) are inherently incapable of maintaining level indefinitely. Throughout this analysis, it is assumed that the HPCI and RCIC pumps cannot pump water at a temperature above 100°C for prolonged periods due to failure of the pump seals. This is purely an assumption, and it is quite conservative. However, the assumption does not have a large impact on the core damage frequency (CDF) for Peach Bottom. Net Positive Suction Head (NPSH) is not an issue as long as the containment remains intact.

This is a Type 2 question, and there are three branches:

1. TL: Operator attempts to control level fail, and the water level is allowed to fall sufficiently that adequate core cooling is not maintained.

2. TAF/OSC: Operator attempts to control level are at least partially successful. Either the level is maintained at Top of Active Fuel (TAF) as specified by the EOPs, or the level oscillates but still provides adequate cooling.

3. na: The sequence is not TC.

CASE # 81:

In this case, the operators failed to inhibit ADS. As the vessel pressure decreased past the interlock setpoint, injection of 20,000 gpm of cold water would begin. Large level and power oscillations would result. Oak Ridge National Laboratory (ORNL) analyzed this sequence [4] for the Browns Ferry Plant (which is very similar to Peach Bottom) and found that containment pressurization will continue at a rapid rate but that core cooling would be maintained as long as water injection continued. The sequence is thus considered to be core vulnerable. This is sequence #61 on the ASEP event tree [3].

QUANTIFICATION

	Optimistic	Central	Pessimistic
TL:	Impossible	Impossible	Impossible
TAF/OSC:	Certain	Certain	Certain
na:	Impossible	Impossible	Impossible

CASE #2:

In this case, ADS is inhibited and SLC is successful. Reactor power is decreasing due to poison injection. ASEP sequences #49 and #50 are covered by this case. Sequence #49 includes a failure to maintain adequate cooling with the low-pressure systems. Sequence #60 includes a failure to depressurize the RPV, preventing injection by these systems, and thus adequate core cooling. In either sequence, water level is not maintained. If it were maintained, this would constitute a success path. The probability that core damage is avoided when SLC is successful is already factored into the TC sequence frequency input to the tree. Thus, this case pertains to the TCUX damage state and failure to maintain adequate level is certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
TL:	Certain	Certain	Certain
TAF/OSC:	Impossible	Impossible	Impossible
na:	Impossible	Impossible	Impossible

CASE #3:

In this case, ADS is inhibited but SLC was not initiated. Vessel pressure was lowered by manual action. The probabilities for Branches 1 and 2 are set to either zero or unity, depending upon which TC sequence is being considered. Branch 3 will always be zero for this case since the case only considers TC. Plant damage states designated as TCSR (X2 or V23) are considered by this case. Since the RPV is depressurized and low-pressure ECCS is available, it is essentially certain that injection will be maintained; at least in the short-term.

QUANTIFICATION

	Optimistic	Central	Pessimistic
TL:	Impossible	Impossible	Impossible
TAF/OSC:	Certain	Certain	Certain
na:	Impossible	Impossible	Impossible

CASE #4:

In this case, no depressurization occurs. The operator is attempting to control the level with only high-pressure systems that are assumed to eventually fail as a result of the high temperatures produced in the suppression pool (from which these systems take suction). The RPV water level is thus sure to fall too low. This sequence is identified on the ASEP ATWS tree as #60. This corresponds to the TCSX plant damage state.

QUANTIFICATION

	Optimistic	Central	Pessimistic
TL:	Certain	Certain	Certain
TAF/OSC:	Impossible	Impossible	Impossible
na:	Impossible	Impossible	Impossible

CASE #5:

This case covers sequences that are not TC. Therefore, this question does not apply.

QUANTIFICATION

	Optimistic	Central	Pessimistic
TL:	Impossible	Impossible	Impossible
TAF/OSC:	Impossible	Impossible	Impossible
na:	Certain	Certain	Certain

QUESTION #19: What type of sequence is this (summary of plant damage)?

This question summarizes the result of the previous 18 questions so as to identify the plant damage state under consideration. It is included for convenience to avoid having to specify the damage state in future questions by long Boolean expressions. The TCSRX2 and TCSRV23 damage states are not distinguished by this question (both are TC-CV) since their distinction depends on the mode of containment failure, which is determined in subsequent questions.

140

This is a Type 2 question, and there are ten branches:

1. AE: The sequence is a large break LOCA with a loss of all injection.

2. AW: The sequence is a large-break LOCA with a loss of the containment heat removal functions of RHR.

3. S1E: The sequence is a small break with a loss of all injection.

4. S1W: The sequence is a small break with a loss of the containment heat removal functions of RHR.

5. TQUV/FTB: The sequence is a transient-initiated loss of all RPV make-up water. Included are station blackouts in which both HPCI and RCIC are inoperable.

6. TB: The sequence is a station blackout in which there is initially injection to the vessel. (The sequence may have been initiated by a plant transient.)

7. TW: The sequence is a transient-initiated event in which core cooling is maintained but the containment heat removal functions of RHR are lost.

8. TC-fDeP: The sequence is a transient followed by a failure to scram (i.e., ATWS). The operator does not depressurize.

9. TC-TL: The sequence is a transient followed by a failure to scram (i.e., ATWS). The operator depressurizes, or fails to inhibit ADS, and allows the level to fall too low to maintain adequate core cooling.

10. TC-CV: The sequence is a transient followed by a failure to scram (i.e., ATWS). The operator controls level at TAF or the level is allowed to oscillate. The CV designation indicates that these are core vulnerable sequences, not necessarily melts.

CASE #1:

In this case, the initiating event is a large-break LOCA. Low-pressure injection systems (LPCI and LPCS) are failed. Because both high-pressure injection systems are turbine driven they are unavailable. This defines the AE sequence.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Certain	Certain	Certain
AW:	Impossible	Impossible	Impossible
S1E:	impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FT8:	Impossible	Impossible	Impossible
TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Impossible	Impossible	Impossible

CASE #2:

A large-break LOCA occurs, followed by a failure of containment heat removal. This defines AW.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Certain	Certain	Certain
S1E:	Impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FT8:	Impossible	Impossible	Impossible
TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Impossible	Impossible	Impossible

CASE #3:

The initiator is a small break. High-pressure injection is terminated by depressurization of the vessel. Low-pressure injection has failed so there is a total loss of all make-up flow. This defines S1E.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Certain	Certain	Certain
S1W:	Impossible	Impossible	Impossible
TQUV/FTB:	Impossible	Impossible	Impossible

TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Impossible	Impossible	Impossible

CASE #4:

A small-break LOCA (in the size range that requires ADS) occurs. Containment heat removal subsequently fails. This defines S1W.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Impossible	Impossible	Impossible
S1W:	Certain	Certain	Certain
TQUV/FTB:	Impossible	Impossible	Impossible
TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Impossible	Impossible	Impossible

CASE #5:

This case covers all transient-initiated events with no injection. This situation defines TQUV. This case includes station blackout with no high-pressure systems available (termed Fast Station Blackout--FTB).

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FTB:	Certain	Certain	Certain
TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Impossible	Impossible	Impossible

CASE #6:

This case includes the remainder of the station blackout sequences (referred to as slow blackouts). Initial operation of at least one of the high pressure systems is implied. Since core cooling can be maintained by the turbine-driven systems for approximately 6 hours (battery depletion prevents further operation), such sequences are termed long-term station blackouts in the tree. Sometimes the term "station blackout" is also used, without qualification, to mean the same type of sequence.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FTB:	Impossible	Impossible	Impossible
TB:	Certain	Certain	Certain
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Impossible	Impossible	Impossible

CASE #7:

In this case, there is a transient initiator followed by a failure of the containment heat removal functions of RHR. This defines TW. For this case, IORV is included as a transient. It should be noted that the large majority of these sequences is expected to be recovered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FTB:	Impossible	Impossible	Impossible
TB:	Impossible	Impossible	Impossible
TW:	Certain	Certain	Certain
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Impossible	Impossible	Impossible

CASE #8:

In this case, the initiator is ATWS, and the operator inhibits ADS and later fails to depressurize. This defines the TC failure-to-depressurize sequence. This is ASEP sequence #50.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FTB:	Impossible	Impossible	Impossible
TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Certain	Certain	Certain
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Impossible	Impossible	Impossible

CASE #9:

In this case, there is an ATWS event. The operator depressurizes the RPV and unsuccessfully attempts to control level with the low-pressure systems. This defines TC failure-to-maintain adequate level sequence. This includes ASEP sequences #59 and #60.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FTB:	Impossible	Impossible	Impossible
TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Certain	Certain	Certain
TC-CV:	Impossible	Impossible	Impossible

CASE #10:

In this case, there is an ATWS and core cooling is at least temporarily maintained using the low-pressure pumps. The containment continues to pressurize. This defines the TC core-vulnerable sequence. For reference, this is ASEP sequence #58.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FTB:	Impossible	Impossible	Impossible
TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Certain	Certain	Certain

CASE #11:

The program should never reach this point since all cases are defined above. However, the program requires an "otherwise" case at the conclusion of the question.

QUANTIFICATION

	Optimistic	Central	Pessimistic
AE:	Impossible	Impossible	Impossible
AW:	Impossible	Impossible	Impossible
S1E:	Impossible	Impossible	Impossible
S1W:	Impossible	Impossible	Impossible
TQUV/FTB:	Impossible	Impossible	Impossible
TB:	Impossible	Impossible	Impossible
TW:	Impossible	Impossible	Impossible
TC-fDeP:	Impossible	Impossible	Impossible
TC-TL:	Impossible	Impossible	Impossible
TC-CV:	Certain	Certain	Certain

QUESTION #20: Is there an LP system break induced by power cycling (ATWS)?

Determines if a break in the low-pressure piping (of LPCS or LPCI) occurs as a result of possible power spikes during an ATWS. This is conceivable since the injection of cold water to the core with little or no reactivity control may occur. This is especially true when the operator is attempting to manually control injection with large capacity systems. Power spikes could be sufficiently severe or rapid that a check valve may fail, resulting in exposure of the low pressure piping to high pressure. It is noted that BWR low-pressure piping is expected to withstand full RPV pressure and has withstood static testing at these pressures. For completeness, allowance has been made for these postulated dynamic failures.

146

Low-pressure system piping in BWRs is designed for approximately 60% of the system operating pressure. Design margins on piping are typically larger than 100%. Thus, one would not expect the low-pressure piping to fail even if it were exposed to operating pressure. Check valves are used in the piping to prevent such exposure, but it is conceivable that these may fail to close or have been inadvertently left open. Indeed, in a recent incident at Browns Ferry, a testable check valve was left in the open position, allowing the low-pressure ECCS (LPCS) piping to be exposed to full RPV pressure. No damage to that piping resulted. Thus, an induced failure of the low-pressure piping is judged to have negligible probability in all cases.

This is a Type 2 question, and there are two branches:

1. PCyBk: A break occurs in the LPCS or LPCI piping as a result of an ATWS power spike.

2. nPCyBk: No break is induced in low-pressure piping as a result of ATWS.

CASE #1:

This case considers ATWS events in which ADS is not inhibited. Large power spikes are expected to accompany ADS. This judgment assumes that no steps were taken to reduce low-pressure injection beforehand (there is little time or reason to do so). Thus it is assumed that both LPCS and LPCI would inject. The voids in the reactor core would collapse. With the core filled with cold water, an over-moderated condition would result. The core power is expected to increase to a level that would be substantially larger than rated power. Fuel rods would heat rapidly, voids would be formed, resulting in the reactor power decreasing. However, with the void formation, there would be an large increase in the vessel pressure. This occurrence is judged to have the largest potential for inducing a pipe break in the low-pressure systems. Oak Ridge National Laboratory analyzed this sequence for the Browns Ferry plant [4] (which is very similar to Peach Bottom) under the Severe Accident Sequence Analysis (SASA) Program. The results of this analysis show that the RPV pressure at no time exceeds the design value (i.e., the SRVs are able to relieve the excess pressure generated). The maximum pressure is thus anticipated to be within the RPV design limit (1350 psia).

QUANTIFICATION

	Optimistic	Central	Pessimistic
PCyBk:	Impossible	Impossible	Impossible
nPCyBk:	Certain	Certain	Certain

CASE2:

This case includes TC sequences in which the operator has limited success in controlling level. There is some opportunity for cycling of the power level to be induced if (or when) the operator depressurizes the vessel and attempts to use the low-pressure, high-capacity systems to control the RPV level. However, since in the sequences considered here, the result of this attempt is that injection is inadequate, there is judged to be little chance of significant power cycles.

QUANTIFICATION

	Optimistic	Central	Pessimistic
PCyBk:	Impossible	Impossible	Impossible
nPCyBk:	Certain	Certain	Certain

CASE #3:

In this case, the operator is maintaining adequate core cooling at a reduced level. Manual control of the ECCS pump flow rates has been taken. It is thus conceivable that a power spike could occur from a sudden increase in water injection. However, it is presumed for these sequences that the operator follows the Emergency Operating Procedures. The logical opportunity for a power spike comes as the vessel is depressurized and injection is transferred to the large-capacity low-pressure pumps. With slow depressurization and adequate throttling of the flow as the injection valves are opened, no problems should result. If the depressurization were too rapid and if the pumps were allowed to begin injection at full flow, a power spike may result as water from the suppression pool (with high subcooling and thus a good moderator) is introduced to the core. With mixing of saturated water already in the vessel any power increase is not likely to be severe enough to cause repressurization of the vessel. A break in the low-pressure piping from repressurization thus induced is even less likely than in the previous cases of this question.

148

QUANTIFICATION

	Optimistic	Central	Pessimistic
PCyBk:	Impossible	Impossible	Impossible
nPCyBk:	Certain	Certain	Certain

CASE #4:

This case covers all sequences that are not TC. Power cycles cannot occur if the Reactor Protection System (RPS) functions. Since there are no power cycles, a power cycle break is impossible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
PCyBk:	Impossible	Impossible	Impossible
nPCyBk:	Certain	Certain	Certain

QUESTION #21: Do low pressure ECC systems continue/begin to inject?

This question determines whether low-pressure systems are used to provide cooling flow to the core. The status of the low-pressure systems (LPCS and LPCI) is also reflected for reference by future questions. This status will be unchanged from that reflected in Question #11 (above) unless the systems were previously available and are now being used for injection.

This is a Type 2 question, and there are four branches:

1. E2fLPC: Low-pressure ECCS (both LPCS and LPCI) have failed.
2. E2rLPC: Neither LPCS nor LPCI are available. However. at least one of these systems is recoverable.
3. E2aLPC: Either LPCS or LPCI (or both) are available (the pumps are running) but are not providing make-up water to the RPV. (Most likely because the RPV pressure is too high.)
4. E2-LPC: Either LPCS or LPCI (or both) are providing make-up water to the RPV.

CASE #1:

In this case, low-pressure systems were initially recoverable and nothing has happened to alter this status. They remain recoverable.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2fLPC:	Impossible	Impossible	Impossible
E2rLPC:	Certain	Certain	Certain
E2aLPC:	Impossible	Impossible	Impossible
E2-LPC:	Impossible	Impossible	Impossible

CASE #2:

In this case, the vessel is not depressurized. The low-pressure systems are available, but there can be no injection since the injection valves are closed. This case includes the TC plant damage states in which the operators fail to depressurize the vessel. Since the case requires that the RPV remain pressurized, it is certain that there will be no low-pressure injection.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2fLPC:	Impossible	Impossible	Impossible
E2rLPC:	Impossible	Impossible	Impossible
E2aLPC:	Certain	Certain	Certain
E2-LPC:	Impossible	Impossible	Impossible

CASE #3:

This case covers two types of sequence. In the first type, both low-pressure systems are failed (LPCS and LPCI) due to hardware failures, and thus they remain failed. The second type of sequence covered by this case is an ATWS in which the operator fails to maintain adequate water level in the core after taking manual control of the RPV level. This possibility was identified by ASEP. The frequency of occurrence for this sequence is also quantified by ASEP. It is noted that the frequency is very low. For this ATWS sequence, adequate low-pressure injection is failed due to operator error.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2fLPC:	Certain	Certain	Certain
E2rLPC:	Impossible	Impossible	Impossible
E2aLPC:	Impossible	Impossible	Impossible
E2-LPC:	Impossible	Impossible	Impossible

CASE #4:

This case covers all sequences in which the vessel is depressurized and at least one of the low-pressure systems is available. Since low pressure injection is automatic for such circumstances, injection is deemed certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E2fLPC:	Impossible	Impossible	Impossible
E2rLPC:	Impossible	Impossible	Impossible
E2aLPC:	Impossible	Impossible	Impossible
E2-LPC:	Certain	Certain	Certain

QUESTION #22: Do containment sprays fail before core degradation?

This question determines the availability of containment spray mode of the RHR system. This includes the availability of spray water only, not the ability to use the sprays for the removal of heat from containment; the latter is considered in questions pertaining to the RHR system. For instance, if Emergency Service Water is failed, the RHR heat exchangers are useless, but spray water can still be directed to the drywell and this would profoundly influence both the accident progression and source terms. Use of HPSW for containment spray has been included. The Peach Bottom P&ID for LPCI indicate that valve alignment to spray service water is possible. While this mode would not remove energy from the containment, the containment temperature and pressure would be lowered. Containment spray is directed when the drywell pressure significantly exceeds the normal level. Interlocks prevent spray operation with the RHR pumps for 10 minutes after a demand for LPCI. Thus, if the RPV level is oscillating, as is likely in an ATWS event, the sprays will shut down as the water level reduces through Level 1 (causing a demand for LPCI). This automatic shutdown of containment spray is judged to reduce the likelihood of continued drywell spray in an ATWS initiated sequence.

This is a Type 2 question. and there are four branches:

1. E3fCS: Containment sprays are failed.

2. E3rCS: Containment sprays are unavailable but may be recovered.

3. E3aCS: Containment sprays are available but are not providing spray to the drywell.

4. E3-CS: Containment sprays are used to spray the drywell.

CASE #1:

In this case, all systems that are capable of supplying drywell spray (RHR/LPCI and HPSW) are failed. Containment spray is thus failed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3fCS:	Certain	Certain	Certain
E3rCS:	Impossible	Impossible	Impossible
E3aCS:	Impossible	Impossible	Impossible
E3-CS:	Impossible	Impossible	Impossible

CASE #2:

This case includes station blackouts in which both systems that are capable of supplying spray (RHR/LPCI and HPSW) are unavailable but can be recovered. Containment spray is thus also unavailable but may be recovered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3fCS:	Impossible	Impossible	Impossible
E3rCS:	Certain	Certain	Certain
E3aCS:	Impossible	Impossible	Impossible
E3-CS:	Impossible	Impossible	Impossible

CASE #3:

In this case, the accident initiator is a primary system break. For such sequences, the drywell pressure is rapidly elevated. The containment spray system is intended to provide rapid reduction in the drywell pressure. In this case, RHR is available; this is assumed to imply that spray mode can be used. That the operator will use spray under

152

these conditions of high containment pressure is judged to be almost certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3fCS:	Impossible	Impossible	Impossible
E3rCS:	Impossible	Impossible	Impossible
E3aCS:	Impossible	Impossible	Impossible
E3-CS:	Certain	Certain	Certain

CASE #4:

In this case, the accident sequence is initiated either by a primary system break or involves a loss of heat removal (TW). The plant damage state is such that RHR is not available, so the use of spray will not lower the total heat within the containment. However, either the RHR or HPSW pumps are available for use in containment spray. (There is a slight inaccuracy here in that the tree references the availability of LPCS and LPCI together so LPCI failed and LPCS available could be the plant damage state for a sequence included in this case. Since none of the ASEP core melt or core-vulnerable cut sets contain this damage state, the event tree results are not impacted.) As noted above, HPSW can be used for containment spray and this will have the effect of lowering the containment temperature and (at least temporarily) the pressure. Since containment spray is not available in the usual sense, and because the use of HPSW requires more unusual operator actions, a lower probability for successful spray is judged to apply in this case than in the previous case, which considers events within the design basis.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3fCS:	Impossible	Impossible	Impossible
E3rCS:	Impossible	Impossible	Impossible
E3aCS:	Impossible	Very Unlikely	Unlikely
E3-CS:	Certain	Very Likely	Likely

CASE #5:

In this case, the accident initiator is an ATWS, and the RHR system is available. Two of the three ASEP TC sequences are considered in this case. The first is that in which the operator initially controls level with HPCI and fails to depressurize the RPV. The second is that in which the water level in the RPV is controlled too low. In either sequence, severe water level oscillations are not expected, and the operator apparently believes the situation is under control. Under such conditions, the operator is all but certain to use the containment spray to reduce containment pressure. Spray will be available, and the containment pressure is high.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3fCS:	Impossible	Impossible	Impossible
E3rCS:	Impossible	Impossible	Impossible
E3aCS:	Highly Unlikely	Very Unlikely	Unlikely
E3-CS:	Highly Likely	Very Likely	Likely

CASE #6:

This case considers the remaining TC plant damage states (those not considered in the previous case). In this sequence, the operators are manually controlling RPV level using LPCS (most likely) or LPCI. It is anticipated that the water level in the RPV will be oscillating. This is expected since the injection capacity of the low-pressure ECCS far exceeds what is needed to maintain adequate water level, and the operator has only rough control of the injection valve position. As described above, water level oscillation around the Level 1 setpoint will cause the spray system to isolate. Thus, the operators are likely to have to attempt repeatedly to initiate spray. The probability that spray will operate for a significant period prior to core damage for this sequence is judged to be indeterminate. The optimistic and pessimistic estimates of the failure probability reflect a reasonable range of uncertainty in this estimate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3fCS:	Impossible	Impossible	Impossible
E3rCS:	Impossible	Impossible	Impossible
E3aCS:	Unlikely	Indeterminate	Certain
E3-CS:	Likely	Indeterminate	Impossible

CASE #7:

This case covers sequences in which the initiating event is a transient and there is little pressurization of the containment because the suppression pool will prevent significant increases in the containment pressure. With no pressurization of the containment, there will be no call for containment spray, so the operator will not use it. Spray remains available, however.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3fCS:	Impossible	Impossible	Impossible
E3rCS:	Impossible	Impossible	Impossible
E3aCS:	Certain	Certain	Certain
E3-CS:	Impossible	Impossible	Impossible

QUESTION #23: Is the containment vented before core degradation?

This question determines whether the operator successfully vents containment prior to core damage. The EOPs instruct containment venting when containment pressure exceeds 60 psig. There are three likely vent pathways: The first is a 2-inch line (with a 1-inch valve) that connects to the torus airspace and is part of the vent and purge system. The second consists of two 18-inch lines, also connected to the torus airspace and also part of the vent and purge system. The third is the Integrated Leak Rate Test (ILRT) line, which connects to the torus and vents outside the reactor building. The latter line has four valves in series (two of which are in the above 18-inch line) and is terminated by a blind flange. The other pathways can be opened from the control room provided ac power is available. For scenarios not initiated by ATWS, deflation of the valve seals on the 18-inch lines probably provides adequate containment pressure control. In addition, there are lines connected to the drywell which could be used for venting. However, since the procedures call for opening the wetwell lines first, and significant time would be required for these actions, the possibility of drywell venting is judged to be negligibly small and is not considered in the containment event tree. Successful venting is defined to mean that a depressurization of the containment occurs. This definition is particularly significant for TC sequences in which opening of the 18" lines is unlikely to allow enough steam to escape the containment to result in the containment pressure being lowered.

This is a Type 2 question, and there are three branches:

1. E3nVENT: The containment is not successfully vented prior to core damage.

2. E3uVENT: Vent valves are opened prior to core damage, but the containment is not depressurized.

3. E3-VENT: The containment is successfully vented prior to core damage.

CASE #1:

This case covers long-term station blackout (high-pressure injection continues until battery depletion). The containment venting pressure of 60 psig will not be reached prior to core damage. It is also true that the operator has no indication of containment pressure due to the loss of ac power, which is used for the pressure transmitters. (A mechanical pressure gauge could be rigged to give this indication, but procedures to do this are not specific.) Because following the emergency procedures would not lead to containment venting in a blackout, the probability of successful venting is assessed to be negligible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nVENT:	Certain	Certain	Certain
E3uVENT:	Impossible	Impossible	Impossible
E3-VENT:	Impossible	Impossible	Impossible

CASE #2:

This case covers break initiators followed by a subsequent loss of containment heat removal. This sequence is slow to progress. Venting (in accordance with EOPs) would occur approximately 8 to 10 hours after the break occurred. The probability of venting for these accidents has been assessed by ASEP. The optimistic and central estimates reflect what is judged to be an appropriate range of this value. It is noted that with a primary system break, there will be alarms indicating high radiation in the containment. These alarms could bring about some reluctance to vent the containment.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nVENT:	Highly Unlikely	Very Unlikely	Unlikely
E3uVENT:	Impossible	Impossible	Impossible
E3-VENT:	Highly Likely	Very Likely	Likely

CASE #3:

In this case, there is a loss of containment heat removal following a transient event. Transient-initiated events are even slower progressing than those that are break initiated. It requires on the order of a day to reach the venting threshold. There will probably be no radiation alarms in this sequence. The operators will have plenty of time to prepare for venting. Public officials can have ample notification to consider measures that may be needed to protect or reassure the surrounding population. For these sequences, venting is the only measure which is expected to avert core damage; thus, it seems all but certain that venting would occur.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nVENT:	Impossible	Highly Unlikely	Unlikely
E3uVENT:	Impossible	Impossible	Impossible
E3-VENT:	Certain	Highly Likely	Likely

CASE #4:

This case covers ATWS initiated sequences in which injection is initially maintained with low-pressure ECCS and is used in two ways: to determine the fraction of the core-vulnerable states which go to core melt and to determine the accident progression once the outcome of the core vulnerable states has been decided. For determining the outcome of the ATWS core-vulnerable states, successful venting is defined to mean arresting the rise in containment pressure. This is doubtful even with four 18-inch vent lines open if the core power level is 18%. However, there are large uncertainties in the power level calculation; this is especially true when the vessel is at low-pressure, as it would be for these sequences. Quantification of the probabilities for venting in this case reflects this uncertainty. It is considered likely that if the operators are successful in opening one of the 18-inch vent lines they will open as many as are required up to the limit of the four that are available. Some uncertainty arises with respect to whether the fact that the opening of the first valve does not reduce the containment pressure is identified by the operators in time to open other lines. Recent calculations performed by Battelle Columbus [8] indicate that with the reactor power level at 12.7% (the current best estimate for low-pressure), two 18-inch lines will maintain the containment pressure below 132 psia. Thus, the issue considered here is whether the operating staff will vent. The procedures do not call for venting until the containment pressure reaches 60 psia. At that time, the containment pressure would be rising rapidly. It requires approximately 20 minutes to jumper the containment isolation system such that the valves can be opened. Considering the rapid pressure rise, successful venting is considered to be unlikely. Quantification of the probability for failure to vent was performed by ASEP

[3]. Their central estimate is 0.9. Their maximum and minimum values are 1.0 and 0.5 respectively. Their assumption is that a maximum entropy distribution is appropriate. In SARRP, the maximum and minimum values are the pessimistic and optimistic quantifications. When this case is used for the purpose of determining the accident progression, the branch point probabilities for this question are either zero or unity. The sequences to be analyzed for accident progression include the phenomenological cut sets in the sequence frequency and thus whether the containment is vented or not has been previously factored into the sequence definition. Sequences involving a power cycle break in the low-pressure systems containment pressure are excluded from this case since the containment is bypassed and no venting will occur.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nVENT:	Certain	Certain	Certain
E3uVENT:	Impossible	Impossible	Impossible
E3-VENT:	Impossible	Impossible	Impossible

CASE #5:

This case covers the two remaining types of TC sequences identified by ASEP. In both these sequences, operator error results in core damage prior to containment failure. The 60-psig venting threshold pressure is unlikely to be reached, and venting initiated, prior to core damage. In both these sequences, the plant operating staff are assumed to have already made errors (in not depressurizing). As in the previous case, the branch point probabilities are either zero or unity depending upon which plant damage state is being considered. When the CET is utilized for the resolution of core vulnerable sequences, branch point probabilities are provided by the ASEP analysts. As in the previous case, power cycle break sequences are excluded from this case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nVENT:	Certain	Certain	Certain
E3uVENT:	Impossible	Impossible	Impossible
E3-VENT:	Impossible	Impossible	Impossible

CASE #6:

This case catches those sequences in which the containment challenge, if any, comes after core degradation begins. Blackouts, in which pressure indication is lost, are not included but have been considered in Case #1 of this question. Since the pressure has not reached the venting threshold, it is assessed to be certain that the operators will not decide to vent. To do so would violate the EOPs.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nVENT:	Certain	Certain	Certain
E3uVENT:	Impossible	Impossible	Impossible
E3-VENT:	Impossible	Impossible	Impossible

QUESTION #24: Does the RPV repressurize before core damage?

The SRVs or ADS valves are held open by air pressure. The valves stay open only as long as the nitrogen pressure in the accumulators (containment isolation isolates the accumulators from instrument nitrogen) exceeds the drywell pressure. If the containment is allowed to pressurize to within 20 to 25 psi of the nitrogen pressure level (i.e., containment at ~110 psi) the valves will reclose and the RPV will begin to repressurize. This will occur regardless of whether the plant staff couples auxiliary nitrogen tanks (which are kept in the reactor building) to the ADS system. Because the nitrogen system has overpressure relief valves, it is simply impossible to overcome containment pressures in the range of the assessed containment failure pressure. This question also considers reclosure of the SRVs due to battery depletion. When the batteries drain, the solenoid valves on the nitrogen accumulators close, subsequent leakage of nitrogen from the valve actuators will result in closure of the SRVs.

This is a Type 2 question, and there are two branches:

1. ReP: The Reactor Pressure Vessel (RPV) repressurizes prior to core damage.
2. nReP: The Reactor Pressure Vessel (RPV) does not repressurize prior to core damage.

CASE #1:

This case excludes TB (station blackout), TW (transient followed by loss of RHR), and TC-CV (ATWS with successful level control). These are the only sequences for which repressurization prior to core damage is possible. This case includes sequences initiated by breaks and transient-initiated sequences where there is a loss of adequate RPV

BEST COPY AVAILABLE

159

water level before containment pressurization is significant. Repressurization does not occur before core damage for these sequences.

QUANTIFICATION

	Optimistic	Central	Pessimistic
RеP:	Impossible	Impossible	Impossible
nReP:	Certain	Certain	Certain

CASE #2:

In this case, there is a stuck-open relief valve. Containment pressurization cannot reseat this valve, so regardless of whether other valves are opened in relief mode and can thus be reclosed, there is judged to be a negligible probability that the RPV could repressurize.

QUANTIFICATION

	Optimistic	Central	Pessimistic
ReP:	Impossible	Impossible	Impossible
nReP:	Certain	Certain	Certain

CASE #3:

In this case, an ATWS power cycle has induced a primary system break in the reactor building. With a break in the primary system, repressurization is impossible. (This case is not accessed in the current tree since induced breaks in the low-pressure piping are assigned negligible probability --Question #20.)

QUANTIFICATION

	Optimistic	Central	Pessimistic
ReP:	Impossible	Impossible	Impossible
nReP:	Certain	Certain	Certain

CASE #4:

In this case, the sequence is a TC, and the operator is successfully controlling the RPV water level using the low-pressure systems. Venting is successful, and the containment pressure decreases. It is certain that the SRVs will not reclose.

QUANTIFICATION

	Optimistic	Central	Pessimistic
ReP:	Impossible	Impossible	Impossible
nReP:	Certain	Certain	Certain

CASE #5:

In this case, the sequence is a long-term blackout (initial high pressure injection). The vessel was depressurized by operator action to stay within the HTCL. Containment pressurization will be insufficient to cause SRV reclosure regardless of any action taken to vent. Repressurization will occur before core damage if the station batteries fail prior to the water make-up pumps (HPCI & RCIC) failing as a result of the pumps overheating. If the batteries fail, injection ceases and the SRVs reclose. If the pumps fail first, the vessel will remain depressurized. Even though there are methods available to the operators for prolonging battery life, battery failure is assumed to occur with high probability. Quantification of the probabilities for repressurization reflects the probability of battery failure for blackout sequences with initial high-pressure make-up. Since Philadelphia Electric's estimate of battery life is 6 to 7 hours under blackout conditions [3] and there are no written procedures available to prolong this period, longer survival is judged very unlikely. A small probability of extended operation is included in the central estimate to reflect uncertainty that could be attributed to the assessment of battery life. This also accounts for the possibility that the operating staff could reduce loads sufficiently to extend the life of the batteries.

QUANTIFICATION

	Optimistic	Central	Pessimistic
ReP:	Likely	Very Likely	Highly Likely
nReP:	Unlikely	Very Unlikely	Highly Unlikely

CASE #6:

This case includes all accident progressions not eliminated by the previous cases in which the RPV has been previously depressurized but the containment has not been vented. These are principally ATWS and loss of containment heat removal scenarios. Continued containment pressurization is certain for such sequences, and thus reclosure of the SRVs is assured.

QUANTIFICATION

	Optimistic	Central	Pessimistic
ReP:	Certain	Certain	Certain
nReP:	Impossible	Impossible	Impossible

CASE #7:

This case covers sequences in which the vessel was not depressurized. Since it was not depressurized, it cannot repressurize. TW sequences in which the containment was successfully vented are also covered by this case. Venting will preclude repressurization.

QUANTIFICATION

	Optimistic	Central	Pessimistic
ReP:	Impossible	Impossible	Impossible
nReP:	Certain	Certain	Certain

QUESTION #25: Does (do) any SRV tailpipe vacuum breaker(s) stick open?

This question determines whether one or more of the vacuum breakers on the SRV tailpipes has stuck open. A stuck-open tailpipe vacuum breaker would provide a pathway for partial bypass of the suppression pool since gases from the tailpipe would pass directly into the drywell. Tailpipe vacuum breakers will open after the associated SRV discharges steam through the tailpipe into the suppression pool. When the steam in the tailpipe condenses on the pipe walls, a vacuum is formed. The vacuum breaker prevents the tailpipe from drawing suppression pool water into it. A stuck-open tailpipe vacuum breaker is only significant if it is the vacuum breaker on the tailpipe for an SRV which is expected to be open after core damage occurs. Thus, the cases below consider which vacuum breakers are challenged by the sequence during the boil-down phase of the accident. The question reflects only significant vacuum breakers being stuck open (i.e., ones that will result in fission product releases bypassing the suppression pool). The probability of a tailpipe vacuum breaker sticking open was estimated by ASEP to be 0.03 per demand, based on available data. For all sequences. this results in a probability of unity that a tailpipe vacuum breaker will fail to reclose. However, the vacuum breakers at Peach Bottom have been modified from the original design (to which the above failure rate applies). The probability of a failure to reclose for the modified design is unknown. Optimistically, it is assumed that the modifications resulted in a factor-of-10 decrease in the failure rate per demand. It is further assumed that there is no reduction in the failure probability for subsequent demands. Reduction of the failure rate for subsequent demand is based on check valve or relief valve performance where the first opening may actually clean the valve seat, thus improving reclosure probability. The

162

BEST COPY AVAILABLE

tailpipe vacuum breakers may actually accumulate damage with subsequent openings; thus, the reduction in the failure rate is judged inappropriate.

This is a Type 2 question, and there are two branches:

1. oSRVBkr: There is a least one SRV tailpipe vacuum breaker stuck open.

2. cSRVBkr: There are no SRV tailpipe vacuum breakers stuck open.

CASE #1:

This case covers transients followed by loss of all injection (TQUV) with no depressurization and no stuck-open relief valves. The vessel will boil down to the point at which core damage occurs in about an hour. Hand calculations (presented in Appendix C) indicate that the boil-off of the RPV inventory of water results approximately 76 lifts of one SRV. The estimated probability of a failure to reclose is thus 0.23 in the central evaluation. The pessimistic evaluation is 1.0, based on assuming the design modifications do not result in significant performance improvement. The optimistic evaluation is a factor-of-10 improvement over the central, a value intended to reflect recognition of the possibility that an SRV may not reclose.

QUANTIFICATION

	Optimistic	Central	Pessimistic
oSRVBkr:	Very Unlikely	Unlikely	Certain
cSRVBkr:	Very Likely	Likely	Impossible

CASE #2:

This case covers long-term isolation transients in which no SRVs stick open. These are similar to scenarios covered in the previous case except that high-pressure systems provide initial core cooling. There are thus more challenges to the tailpipe vacuum breakers. With the vessel at high pressure (with no SORV), the low-low set SRV will cycle. This is the same valve through which fission products could be released if core damage followed; thus, a stuck-open vacuum breaker could be significant. Since hand calculation of the number of SRV lifts is very crude, no distinction is made in quantifying the branch point probabilities for this case and the previous case. This approach simplifies the treatment of uncertainties with regard to this issue in the Limited Latin Hypercube sampling of the event tree.

QUANTIFICATION

	Optimistic	Central	Pessimistic
oSRVBkr:	Very Unlikely	Unlikely	Certain
cSRVBkr:	Very Likely	Likely	Impossible

CASE #3:

This case includes TC sequences with a failure to depressurize the vessel. The power level is such that one or more SRVs will be held open in relief mode if depressurization does not occur. The valve that would cycle, thereby challenging the tailpipe vacuum breaker, is not the low-low set valve and thus would not be the valve through which fission products pass. A very small probability of this bypass mode is assigned in the central estimate of this case to recognize the possibility. Challenges to the low-low set valve might occur late in the sequence. The optimistic and pessimistic estimates reflect a factor-of-10 variation in central estimate probability of a failure to reclose a tailpipe vacuum breaker causing a partial suppression pool bypass.

QUANTIFICATION

	Optimistic	Central	Pessimistic
oSRVBkr:	Impossible	Very Unlikely	Unlikely
cRVBkr:	Certain	Very Likely	Likely

CASE #4:

This case covers sequences with primary system breaks and/or early vessel depressurization. For such cases, the release pathway is not through an SRV that would have been cycling earlier. Thus, there can be no bypass through any stuck open tailpipe vacuum breaker.

QUANTIFICATION

	Optimistic	Central	Pessimistic
oSRVBkr:	Impossible	Impossible	Impossible
cSRVBkr:	Certain	Certain	Certain

QUESTION #26: Is there a repressurization break of low-pressure systems?

Determines whether a break occurs in the low-pressure piping of the LPCS or LPCI systems as a result of the RPV repressurization. Since the low-pressure piping on a BWR can probably withstand primary system pressure, and since the check valves should function to protect the "low-pressure" piping, a break induced as a result of the vessel repressurizing is not considered to be of significant probability. This question is included primarily for completeness since earlier BWR risk assessments have shown "V" sequences to be significant risk contributors. While ASEP did not identify interfacing system LOCAs as significant initiating events, they may be considered more likely if the vessel is repressurized with the low-pressure ECCS operating. In this case, action of the check valve, exclusive of pressure interlocks closing the injection valves, is needed to protect the upstream piping from full RPV pressure. However, even if these valves were not to close, it is judged to be of negligible probability that the piping would fail. Question #20 (above) has more discussion of this issue.

This is a Type 2 question, and there are two branches:

1. RePBk: A break in the low-pressure ECCS piping results from RPV repressurization.

2. nRePBk: There are no breaks in the low-pressure ECCS piping resulting from vessel repressurization.

CASE #1:

This case includes all sequences in which there is a repressurization of the primary system prior to core damage (determined by Question #24). RPV pressure thus increases past the shutoff point of low-pressure injection. Only in such sequences is a repressurization break possible. Quantification reflects the probability that the check valve will fail to reclose and that the piping has some defect that results in a break occurring. The piping has sufficient design strength to withstand the pressure; thus, the failure probability is taken as negligible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
RePBk:	Impossible	Impossible	Impossible
nRePBk:	Certain	Certain	Certain

CASE #2:

This case includes all sequences where there is no repressurization of the RPV. Therefore, a repressurization break is impossible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
RePBk:	Impossible	Impossible	Impossible
nRePBk:	Certain	Certain	Certain

QUESTION #27: Does containment fail before core degradation?

This question determines whether or not the containment fails prior to core degradation. This is possible only when injection to the vessel is maintained but heat removal from the containment is inadequate. This question refers explicitly to failure of containment. Breaks or leakage in systems interfacing with the RPV are not considered containment failures but are accounted for in subsequent questions.

This is a Type 2 question, and there are two branches:

1. E3-CFP: The containment fails as a result of overpressurization prior to core degradation.
2. E3nCFP: The containment is not failed prior to core degradation.

CASE #1:

This case oovers sequences with an induced break in the low-pressure ECCS piping (Question #20 or Question #26). With such a break, containment is bypassed. Since there is no challenge to the containment, overpressure failure is impossible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-CFP:	Impossible	Impossible	Impossible
E3nCFP:	Certain	Certain	Certain

CASE #2:

In this case, the containment is successfully vented. By definition, this precludes containment failure due to overpressure. For this case, failure is then impossible.

166

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-CFP:	Impossible	Impossible	Impossible
E3nCFP:	Certain	Certain	Certain

CASE #3:

This case includes all sequences in which there is no challenge to containment prior to core degradation. Included are all sequences without initial injection to the vessel, blackouts, and ATWS events in which injection is insufficient to maintain adequate core cooling even though the ECC systems are operable (failure-to-depressurize or level-maintained-too-low sequences). For all these sequences, core damage will occur before the pressure rise in the containment is sufficient to challenge its integrity. Loss of containment heat removal (TW) sequences and the core-vulnerable TC sequences are specifically excluded. The latter sequences are considered individually in the remaining cases of this question since distinctions must be drawn when the CET is used for the resolution of ASEP cut sets resulting in the core vulnerable condition.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-CFP:	Impossible	Impossible	Impossible
E3nCFP:	Certain	Certain	Certain

CASE #4:

In this case, the sequence is an ATWS with adequate core cooling being provided by the low-pressure systems. The power level is estimated to be 13% to 18%. Venting was not attempted or was unsuccessful, and the containment pressurizes rapidly. As the containment continues to pressurize, there are two possibilities. Either the containment will fail or the SRVs will reclose due to the elevated containment pressure early enough to prevent pressurization of the containment to the point of failure. In the latter case, core degradation will occur in an intact containment. The branch point probabilities for this case thus determine if containment fails after injection is lost but before the level in the RPV boils down to the point at which core damage will occur. The probability of containment failure was addressed with a simple hand calculation assuming 470,000 pounds of water are boiled off to the pool after SRV reclosure prior to core damage. That calculation (in Appendix C) demonstrates that the containment pressure would certainly be raised far beyond the mean failure pressure. However, the calculation assumed water level to be maintained at the normal level. This is not the expected case. Rather one expects that the level would oscillate between the normal level and TAF. Thus, the probability that the containment will remain intact is uncertain; it depends on the water inventory of the vessel at the time the vessel pressure reaches the cut-off pressure for low-pressure injection. In the analysis of accident progression, branch point probabilities will be zero or unity depending upon the group of cut sets being considered. The cut sets include containment failure as a base event, and thus the cut sets determine the branch point probabilities for this case. In assessing the sequence frequency for these cut sets (i.e., when the CET was used to resolve the core-vulnerable sequences), the containment failure probability for this case was 0.6. This quantification reflects the authors' judgment that water level in the RPV is likely to be maintained near TAF, and thus, once injection fails, there is a significant probability that the containment will not overpressurize.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-CFP:	Certain	Certain	Certain
E3nCFP:	Impossible	Impossible	Impossible

CASE #5:

This case includes transient-initiated sequences with loss of containment heat removal. High pressure systems will fail on high containment pressure (this case includes no venting) or high pool temperature, the RPV will repressurize eliminating low pressure injection. CRD flow is also not available so there is no injection. Unlike the previous case, at the time injection fails, the RPV water level will be in the normal range. As indicated by the calculation described for that case (Appendix C), containment failure is certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-CFP:	Certain	Certain	Certain
E3nCFP:	Impossible	Impossible	Impossible

QUESTION #28: What is location of early containment leakage?

This question determines whether early containment failure occurs in the wetwell or in the drywell. Given that the containment fails (determined in the previous question), the split fraction between wetwell and drywell was estimated based on the weighting factors placed on various modes of containment failure by a group of experts assembled by SARRP to consider risk significant issues for Peach Bottom [6] (see Appendix C). Containment leakage, induced by venting, isolation failure, or breaks in interfacing systems are included in this question.

This is a Type 2 question, and there are three branches:

1. E3nCL: There is no leakage from the containment prior to core degradation, that is in excess of technical specification limits.
2. E3-CLD: There is containment leakage from the drywell prior to core degradation.
3. E3-CLW: There is containment leakage from the wetwell prior to core degradation.

CASE #1:

In this case, there has been an induced LOCA in the low-pressure ECCS piping. The RPV is thus open to the reactor building. While the containment structure is intact, there is leakage from the containment and it originates in the drywell.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CLD:	Certain	Certain	Certain
E3-CLW:	Impossible	Impossible	Impossible

CASE #2:

In this case, the containment has been vented. Leakage is thus from the wetwell since drywell venting is assumed to be of negligible probability. This case includes both successful and unsuccessful venting (defined in Question #23). In either situation, containment leakage will be predominantly from the torus (wetwell).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CLD:	Impossible	Impossible	Impossible
E3-CLW:	Certain	Certain	Certain

CASE #3:

This case covers only ATWS-initiated sequences in which containment failure followed successful injection with low-pressure ECCS. These are referred to as the TC-core vulnerable (TC-CV) sequences. In the CET evaluation used to resolve the subset of these sequences that resulted in core damage, this case was quantified exactly as is the one which follows it. However, in the analysis of the accident progression that follows from a plant damage state (i.e., given that core damage would occur), this case must be quantified so as to preserve the branch point quantification for which is appropriate for the sequences determined to lead to core damage. Thus, the quantification below applies for the TCSRX2 damage state. If the TCSRV23 damage state is analyzed by the tree, a different quantification, involving a much larger probability of torus failure, is used.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CLD:	Indeterminate	Very Likely	Certain
E3-CLW:	Indeterminate	Very Unlikely	Impossible

CASE #4:

There has been an overpressure failure of the containment. The probability of drywell (versus wetwell) failure is estimated from the expert responses (Appendix C). The optimistic estimate is the highest weighting assigned to wetwell failure by any one expert. The pessimistic estimate corresponds to the majority of experts who gave no weight to wetwell failure.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CLD:	Indeterminate	Likely	Certain
E3-CLW:	Indeterminate	Unlikely	Impossible

CASE #5:

This case includes all sequences in which the containment is not breached prior to core degradation. The probability is thus unity that the containment does not leak.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Certain	Certain	Certain
E3-CLD:	Impossible	Impossible	Impossible
E3-CLW:	Impossible	Impossible	Impossible

QUESTION #29: For drywell failures, is the failure at the head seal?

This question applies to sequences in which early containment failure occurs in the drywell (determined by the previous question). Two possible drywell failure modes have been identified. The Ames Laboratory analysis [7] concluded that the weakest point in the drywell wall was at the knuckle (junction between cylindrical and spherical sections). However, leakage at the head seal is also possible. Leakage flows from the latter location pass directly into the refueling bay, bypassing the reactor building. If the reactor building is bypassed, there may be a significant reduction in the retention of fission products by the secondary containment. The split fraction between failure at the head seal and locations lower in the drywell (represented by the knuckle location) was determined on the basis of the same expert opinions described in the previous question. (See Appendix C.)

This is a Type 2 question, and there are three branches:

1. E3-DHF: There is leakage from the drywell head seal prior to core degradation.

2. E3-DKF: There is leakage from the drywell to the reactor building (from the knuckle) prior to core degradation.

3. E3nDF: The drywell retains its integrity prior to core degradation.

CASE #1:

In this case, there has been an induced failure of the low-pressure ECCS piping in the reactor building. The leakage pathway from inside the containment is to the reactor building. This is assumed equivalent to a knuckle failure.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-DHF:	Impossible	Impossible	Impossible
E3-DKF:	Certain	Certain	Certain
E3nDF:	Impossible	Impossible	Impossible

CASE #2:

This case applies to ATWS sequences that were among those originally determined by ASEP to result in a core-vulnerable condition (i.e., containment failure occurs and it is uncertain as to whether injection to the RPV can be maintained). Analysis with the CET was used to determine which sequences did not lead to core melt. For that purpose, the case was quantified in a manner identical to the one that follows it. When the CET was employed to analyze the accident progression pathways that evolve given a particular plant damage state (i.e., conditional probability of core damage is unity), the quantification of this question is changed to be consistent with the preconditions of the damage state. The quantification shown in Appendix E is that which is appropriate for the TCSRX2 damage state. This state presumes containment leakage (Level 2--defined in Question #30) prior to core damage. Thus, the probability that the failure location is the drywell head seal is much larger than for the case in which the leakage level is not defined by the damage state (i.e., damage states which do not derive from ASEP core-vulnerable sequences). The larger probability of head seal failure derives from the assumption of the containment performance experts that head seal failure results only in leakage and will not lead to rapid depressurization of the the containment. (Thus, given that the containment leaks, it is far more likely that the leak is from the drywell head.) When this case is applied to the TCSRV23 damage state, a zero probability of head seal leakage applies since this damage state presumes a rupture of the containment prior to core damage.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-DHF:	Unlikely	Likely	Certain
E3-DKF:	Likely	Unlikely	Impossible
E3nDF:	Impossible	Impossible	Impossible

172

CASE #3:

In this case, the drywell has been assumed to have failed due to overpressurization by steam. The central estimate of the split fraction between the head seal leakage location and locations lower in the drywell (represented by the knuckle) was determined from the average of the expert weightings. Since the result is close to a 50-50 split, the optimistic and pessimistic evaluations were taken as the typical range assumed in this event tree around an indeterminate value (i.e., 0.1 and 0.9).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-DHF:	Unlikely	Indeterminate	Likely
E3-DKF:	Likely	Indeterminate	Unlikely
E3nDF:	Impossible	Impossible	Impossible

CASE #4:

This case covers sequences in which the drywell retains its integrity. The probability of no drywell leakage is thus unity.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3-DHF:	Impossible	Impossible	Impossible
E3-DKF:	Impossible	Impossible	Impossible
E3nDF:	Certain	Certain	Certain

QUESTION #30: What is the containment leakage level before core degradation?

This question determines the level (or rate) of leakage from the containment prior to core degradation. Two distinct levels of leakage are allowed. Level 2 corresponds to a small leak that provides essentially no reduction in the containment pressure (leak rate approximately equal to vessel steaming rate). Level 3 corresponds to a rupture of the containment (the containment depressurizes). The probabilities for the various levels were determined based on weights assigned by the SARRP expert review group [6] to these levels. considering various modes of containment failure. (See Appendix C.)

This is a Type 2 question, and there are three branches:

1. E3nCL: There is no leakage from the containment significantly in excess of technical specifications prior to core damage.

2. E3-CL2: There is leakage from the containment prior to core damage. However, the leakage is insufficient to result in depressurization of the containment.

3. E3-CL3: There is a failure of containment prior to core damage of a size sufficient to cause depressurization of the containment.

CASE #1:

In this case, a break has been induced in the low pressure ECC system piping. The containment is effectively bypassed. Containment pressure will remain at approximately its current level so long as the RPV remains intact. However, leakage of steam from the vessel is sufficiently large that, with respect to impact on the reactor building, this is considered to be Level 3 leakage. This classification is consistent with subsequent references to this question.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CL2:	Impossible	Impossible	Impossible
E3-CL3:	Certain	Certain	Certain

CASE #2:

This case covers scenarios in which one or more containment vents were opened, but this action did not result in depressurization of the containment. This case would thus apply only to ATWS sequences in which venting was attempted but sufficient flow did not escape the valves in the vent line(s). Since there would be leakage from the containment in this case, and the containment would not depressurize (by definition of unsuccessful venting--Question #23), this situation fits the definition of Level 2 leakage.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CL2:	Certain	Certain	Certain
E3-CL3:	Impossible	Impossible	Impossible

174

CASE #3:

In this case, the containment has been successfully vented. With the containment vented, leakage from the containment is, by the definition of successful venting. Level 3.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CL2:	Impossible	Impossible	Impossible
E3-CL3:	Certain	Certain	Certain

CASE #4:

This case covers the ATWS plant damage states that derive from the sequences originally determined by ASEP to be core vulnerable. The discussion under Case #2 of Question #29 is appropriate. To resolve core-vulnerable sequences, this case was quantified exactly as is the succeeding case. When the CET is used for the analysis of the accident progression that follows from a given plant damage state, the quantification of this case reflects the damage state (either TCSRX2 or TCSRV23) definition. For TCSRX2, Level 2 leakage is certain. For TCSRV23, Level 3 leakage is certain. The quantification for TCSRX2 is used in this example.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CL2:	Certain	Certain	Certain
E3-CL3:	Impossible	Impossible	Impossible

CASE #5:

This case is not relevant to the current analysis. It is included to allow treatment of ATWS-initiated sequences not in the current core damage profile. Such sequences would have to have containment failure prior to core damage. The failure is assumed to be Level 3, as this case is currently quantified.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CL2:	Impossible	Impossible	Impossible
E3-CL3:	Certain	Certain	Certain

175

CASE #6:

This case covers scenarios (not initiated by ATWS, since these are considered in the two previous cases) in which containment leakage occurs prior to core damage. The split between leakage (Level 2) and rupture (Level 3) was determined on the basis of the average of the weights assigned to these two levels by the SARRP containment performance expert review group [6]. This case presumes that the failure location was not the drywell head seal. Head seal failure is considered in the succeeding case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CL2:	Likely	Unlikely	Impossible
E3-CL3:	Unlikely	Likely	Certain

CASE #7:

In this case, early failure occurs at the head seal. There was general consensus among the experts polled that this failure location would always result in a leak. The tortuous leakage pathway around the seal was cited as justification for this conclusion. Due to the consensus between the experts polled by SARRP and the apparent similar conclusion by IDCOR, the quantification for all walkthroughs is the same.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Impossible	Impossible	Impossible
E3-CL2:	Certain	Certain	Certain
E3-CL3:	Impossible	Impossible	Impossible

CASE #8:

In this case, the containment is not bypassed, vented, or failed. The probability of significant leakage is thus negligible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E3nCL:	Certain	Certain	Certain
E3-CL2:	Impossible	Impossible	Impossible
E3-CL3:	Impossible	Impossible	Impossible

QUESTION #31: Is the suppression pool drained before CD?

If the accident progression involves failure of the torus (wetwell), the failure may occur below the level of the pool. If this were the case, the pool would begin to drain. This question considers this possibility. The three branches of this question consider differing levels of severity for this failure. Branch 1 presumes that the failure results in the suppression pool being drained almost entirely, to the point that there is essentially no water covering the SRV quenchers. This is extremely unlikely to occur at Peach Bottom since the torus room is watertight and would retain water leaked from the pool. The torus room would thus prevent the water level in the pool from falling lower than some level at which the level in the torus room would equal the level remaining in the pool. This situation is identified as Branch 2. Branch 2 presumes that the drywell vents are essentially uncovered but that the SRV quencher discharges are essentially unaffected by the loss of level. Branch 3 corresponds to no failure in the region of the torus beneath the water level. In the quantification of this question, consideration is given to the fact that the torus at Peach Bottom has a thicker wall in that region, which is below the water line. However, the added thickness of steel is little more that 10% (0.676 versus 0.604 inch). This added thickness is not believed to lend sufficient extra strength to assure that any failure would be above the water line. Specifically, local stresses around stiffners and penetrations would likely dominate the failure modes of the torus.

This is a Type 2 question, and there are three branches:

1. E4-SPD: The suppression pool drains prior to core damage such that there is complete bypass.
2. E4pSPD: The suppression pool level is reduced by leakage prior to core damage. The drywell vents are uncovered and consequently represent a suppression pool bypass pathway.
3. E4nSPD: The suppression pool water inventory is maintained.

CASE #1:

This case represents the only circumstances under which drainage from the suppression pool is possible: a rupture of the containment occurs in the torus (wetwell). Accounting for the added thickness of steel below the water line leads to the assessed 60-40 split that the failure does not result in any drainage. Given that failure occurs below the water line, a 10% probability is assumed that the torus room is not watertight. Leakage of water from the torus room could result either from the water tight door being left open inadvertently or from a failure of the door resulting from the sudden imposition of the large loadings that would accompany failure of the torus.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-SPD:	Impossible	Very Unlikely	Unlikely
E4PSPD:	Very Unlikely	Indeterminate	Likely
E4nSPD:	Very Likely	Indeterminate	Unlikely

CASE2:

This case covers sequences in which there is either no containment failure prior to core damage, failure occurred in the drywell, or there is only a Level 2 leak in the torus that is assumed too small to lower the water level significantly during core degradation.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-SPD:	Impossible	Impossible	Impossible
E4PSPD:	Impossible	Impossible	Impossible
E4nSPD:	Certain	Certain	Certain

QUESTION #32: Does ac power remain lost during core degradation?

This question accounts for the delayed recovery of offsite Power. ASEP accounts for recovery that occurs prior to core damage in estimating the sequence frequency, but later recovery is also possible. Late recovery of ac may prevent vessel meltthrough and otherwise mitigate accident consequences. The probability of nonrecovery for this question is estimated from the generic grid data used by ASEP. The optimistic and pessimistic estimates represent, respectively, the 5th and 95th percentile of the available offsite power recovery data. These data are the basis of NUREG-1032. A correction factor was applied to the nonrecovery probabilities derived from the data such that the mean value (the central estimate) corresponded to the value that would be derived based on the information presented in NUREG-1032.

This is a Type 2 question. and there are two branches:

1. E4fAC: Ac power is not recovered prior to vessel breach.

2. E4-ac: Ac power is recovered during core degradation.

CASE #1:

This case includes blackouts in which there is early high pressure coolant make-up. AC power would have to be restored between 6 and 8 hours after the blackout to prevent damage; vessel failure may be prevented with recovery by 11 hours. The recovery probability was estimated based on sequence timing results from the Source Term Code Package (STCP) calculations [9]. These calculations show core uncovery at 8.8 hours and core slump at 11.6 hours. The nonrecovery probability for this case was estimated assuming that power must be restored by 11.6 hours to be efficacious. This case presumes that high-pressure injection fails as a result of the depletion of the station batteries. Thus, offsite power must be recovered in the absence of dc power. ASEP assumed (for the TBUX plant damage state) that offsite power could not be recovered without dc power. In this analysis, it is assumed that recovery is more difficult without dc power since switches and circuit breakers must be reset manually. An increased probability of nonrecovery in this case is estimated by multiplying the nonrecovery probabilities by the ratio of the 96th percentile value to the mean.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fAC:	Unlikely	Indeterminate	Indeterminate
E4-ac:	Likely	Indeterminate	Indeterminate

CASE #2:

This case is similar to the previous one except that high pressure injection is assumed to fail as a result of high suppression pool temperature rather than battery depletion. DC power is thus available for the resetting of switches and circuit breakers. The timing of the scenario is assumed to be the same as that for CASE #1. The quantification of the central estimate approximates the results which would be obtained from NUREG 1032.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fAC:	Unlikely	Indeterminate	Indeterminate
E4-ac:	Likely	Indeterminate	Indeterminate

CASE #3:

This case covers sequences in which the loss of offsite power is followed by failure of the diesel generators and independent failure of HPCI and RCIC (i.e.. TBU plant damage state). For this damage state, ASEP considered recovery of power between 30 and 40 minutes following the loss of offsite power. The quantification of this case assumes that power must be recovered within two hours of the blackout in order to have any chance of preventing vessel breach (this timing is based roughly on STCP calculation Performed for Grand Gulf 10). Thus the central estimate represents the power nonrecovery probahility at 1.8 hours, given that no power was available after 36 minutes, based on the information in NUREG-1032.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fAC:	Indeterminate	Indeterminate	Likely
E4-ac:	Indeterminate	Indeterminate	Unlikely

CASE 4:

This case covers the TBUX plant damage state in which loss of offsite power is followed by common mode failure of the station batteries. The timing of the progression of this accident is similar to that of the previous case. However, since dc power is failed, the nonrecovery prohability will be increased. ASEP assumed a nonrecovery probability of unity for this case. To reflect the decrease of offsite power recovery probability of this case over the previous one, the nonrecovery probabilities were multiplied by the ratio of the 95th percentile value to the mean.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fAC:	Indeterminate	Likely	Very Likely
E4-ac:	Indeterminate	Unlikely	Very Unlikely

180

CASE #5:

For the remaining sequences, some form of ac power is available (either emergency or offsite); thus, ac power remains available with a probability of unity.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fAC:	Impossible	Impossible	Impossible
E4-ac:	Certain	Certain	Certain

QUESTION #33: What is the status of CRD?

This question determines the status of the control rod drive (CRD) system. Recovery due to the restoration of offsite power and the possible negative impact of reactor building steam flooding are considered. Steam flooding of the reactor building is a possible consequence of containment failure or venting. This occurrence may impact the operability of the CRD pumps by causing shorts or malfunctions in the motor controls. At Peach Bottom, unlike the majority of Mark I and Mark II containment design BWRs, the CRD pumps are in the turbine building rather than the reactor building. It is thus our assessment that containment failure (or venting) will have a negligible impact on CRD pump operability.

This is a Type 2 question, and there are three branches:

1. E34fCRD: The CRD cooling flow is failed.

2. E34rCRD: The CRD cooling flow is unavailable, but may be recovered.

3. E34-CRD: There is flow from the CRD cooling system into the vessel.

CASE #1:

This case includes sequences in which the containment ruptures (or is vented) as a result of overpressurization by steam (station blackouts with early venting are excluded). The pool is thus superheated, and large volumes of steam would be released. Alternatively, there has been an induced LOCA in the low-pressure ECCS piping. In either case, the reactor building will be flooded with steam but since there will be little leakage to the turbine building, the CRD pumps are judged certain to continue to operate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fCRD:	Impossible	Impossible	Impossible
E34rCRD:	Impossible	Impossible	Impossible
E34-CRD:	Certain	Certain	Certain

CASE2:

In this case, the CRD pumps are not currently injecting but are recoverable. Thus, this case is similar to Case 1 of this question except that the CRD flow remains recoverable rather than remaining operational.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fCRD:	Impossible	Impossible	Impossible
E34rCRD:	Certain	Certain	Certain
E34-CRD:	Impossible	Impossible	Impossible

CASE #3:

In this case, CRD pumps have been providing injection all along or have been recovered by the recovery of ac power. The containment is intact or has only Level 2 leakage. The pumps are considered certain to remain operational.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fCRD:	Impossible	Impossible	Impossible
E34rCRD:	Impossible	Impossible	Impossible
E34-CRD:	Certain	Certain	Certain

CASE4:

In this case, there is also no significant leakage to the containment to impact the status of the CRD cooling system. The system was unavailable but recoverable. it remains in this condition.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fCRD:	Impossible	Impossible	Impossible
E34rCRD:	Certain	Certain	Certain
E34-CRD:	Impossible	Impossible	Impossible

CASE #5:

This case includes only those sequences in which the CRD cooling system was previously failed. This being the case, it is certain to remain failed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fCRD:	Certain	Certain	Certain
E34rCRD:	Impossible	Impossible	Impossible
E34-CRD:	Impossible	Impossible	Impossible

QUESTION #34: What is the status of HPSW?

This question determines the availability of High-Pressure Service Water for injection to the vessel through the cross-tie with the LPCI system. Operator action is required to open the valves on the cross-tie line. To inject to the vessel, the LPCI injection valves must be opened (preferably shutdown cooling mode). The HPSW pumps require offsite power. The HPSW pumps are not in the reactor building and thus could not fall victim to induced failure by steam flooding. However, the motor-operated valves on the cross-tie are in the reactor building and are therefore vulnerable. Since these valves are normally closed, failure of the valves would result in HPSW being unavailable for injection to the RPV. Considerations in quantifying this question are similar to those in the previous question. While the valve operators are probably better able to withstand severe environments than motor controls (recent qualification problems with motor operated valves for BWRs are probably not relevant), the valves are more likely to be exposed to the severe environment (and for a longer period). On balance, the authors' judgment is that failure is unlikely.

This is a Type 2 question, and there are three branches:

1. E34fHSW: High-Pressure Service Water has failed in a manner that precludes injection to the vessel.
2. E34rHSW: High-Pressure Service Water is unavailable for injection to the vessel but is recoverable.
3. E34aHSW: High-Pressure Service Water is available to supply low-pressure injection to the RPV.

CASE #1:

In this case, as in Case 1 of the previous question, there is a high level of steam flooding in the reactor building. Prior to the loss of containment integrity, either HPSW was available or it was going to be recovered through the restoration of ac power. Unlike safety-related systems, the HPSW cross-tie is not qualified for severe conditions in the reactor building. The probability that the HPSW cross-tie will be inoperable, and thus HPSW will be unavailable (for either vessel injection or containment spray), is judged to be significant.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fHSW:	Highly Unlikely	Unlikely	Indeterminate
E34rHSW:	Impossible	Impossible	Impossible
E34aHSW:	Highly Likely	Likely	Indeterminate

CASE #2:

This case is similar to Case 1 of this question, except that HPSW was unavailable prior to containment failure. While the condition was recoverable, steam flooding of the reactor building could induce failure. Quantification of the branch probabilities for this case is exactly as in Case 1 with the exception that even if there is no induced failure, HPSW is only recoverable rather than judged to be available for injection to the vessel.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fHSW:	Highly Unlikely	Unlikely	Indeterminate
E34rHSW:	Highly Likely	Likely	Indeterminate
E34aHSW:	Impossible	Impossible	Impossible

CASE #3:

In this case, either HPSW is available and thus remains so, or HPSW was recoverable and ac power has been recovered, making HPSW once more available. Any leakage (Level 2 or less) that may be occurring from the containment to the reactor building is judged to be insufficient to impact valve operability.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fHSW:	Impossible	Impossible	Impossible
E34rHSW:	Impossible	Impossible	Impossible
E34aHSW:	Certain		

CASE #4:

In this case, HPSW was initially recoverable. Leakage from the containment is sufficiently low (Level 2 or less) that it is judged that there would be no impact on valve operability. It is thus certain that HPSW will remain recoverable.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fHSW:	Impossible	Impossible	Impossible
E34rHSW:	Certain	Certain	Certain
E34aHSW:	Impossible	Impossible	Impossible

CASE #5:

This case catches all sequences where HPSW is initially failed. This being the case it certainly remains failed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E34fHSW:	Certain	Certain	Certain
E34rHSW:	Impossible	Impossible	Impossible
E34aHSW:	Impossible	Impossible	Impossible

QUESTION #35: Does the core melt?

This question resolves core-vulnerable sequences into two groups: (1) those which end in safe stable conditions and (2) those in which core damage occurs. All sequences defined by ASEP to be core melts are properly identified as such by this question.

This is a Type 2 question, and there are two branches:

1. MELT: At least some core damage occurs.
2. nMELT: Core vulnerable sequences assigned to this branch result in stable shutdowns with no core damage.

CASE #1:

This case covers sequences initiated by a primary system break in which the containment heat removal system fails. However, the condensate system is available and could inject to the vessel. Operator action is required to align the condensate system and restart the condensate pumps. It is assumed that given that the operator does this, core damage will be avoided. (Even if no venting occurs, the containment would flood.) Quantification of the probabilities for this case is based on the fact that the condensate system is one with which the operator is very familiar. Condensate would, more than likely, be the system the operator would turn to. Given this, it is the authors' judgment that the use of condensate (given the 6 to 10 hours available) is highly likely. A factor of 100 uncertainty in the failure probability is applied for the optimistic and pessimistic estimates.

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELT:	Impossible	Highly Unlikely	Unlikely
nMELT:	Certain	Highly Likely	Likely

CASE #2:

This case is similar to Case 1 of this question, the difference being that instead of condensate being available for injection to the vessel, it is HPSW. This case is quantified in exactly the same manner. It is recognized that use of the HPSW cross-tie is to a familiar action for the operator. However, in these sequences, containment pressurization is sufficiently slow that the operator would have on the order of ten hours to align HPSW. It is highly unlikely that the operator would allow the core to melt with any injection system that is identified in the procedures as being available, given this much time.

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELT:	Impossible	Highly Unlikely	Unlikely
nMELT:	Certain	Highly Likely	Likely

CASE #3:

This case is again similar to Cases 1 and 2 of this question. In this case, however, neither condensate or HPSW is available for injection. The only system that is available long-term is the CRD cooling system. For these sequences the containment must be vented or eventually failed. It is assumed that this will cause loss of Net Positive Suction Head (NPSH) for the ECCS pumps. Calculations [11] by Oak Ridge National Laboratory (ORNL) for Brown's Ferry (similar to Peach Bottom) indicate that CRD flow can remove decay heat from the core 6 to 8 hours after shutdown. These calculations were performed for a transient initiator. It was also assumed that the operators took action to optimize the flow from the CRD pumps. There is some uncertainty that CRD flow could be adequate when there is a primary system break. This is particularly true if the elevation of the break is below Top of Active Fuel (TAF). This uncertainty is accounted for in the evaluation of the success probabilities for this case. The central estimate is that CRD flow is likely to succeed. The more likely breaks are small, and the more likely locations for larger breaks are above the top of the core; thus, the more likely sequences for this case are similar to the transients analyzed by ORNL. The pessimistic estimate is that the success probability is indeterminate, reflecting the lack of analysis directly applicable to this case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELR:	Highly Unlikely	Unlikely	Indeterminate
nMELT:	Highly Likely	Likely	Indeterminate

CASE #4:

In this case, the initiating event is a transient. Referring to the comments in Case 3 of this question, the Oak Ridge calculations indicate that CRD flow will provide adequate cooling for long-term transients. This being the case, the probability of core melt is negligible for a TW sequence with CRD flow available, regardless of RPV pressure.

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELT:	Impossible	Impossible	Impossible
nMELT:	Certain	Certain	Certain

CASE #5:

In this case, the sequence was initiated by a transient, and containment heat removal subsequently failed (TW). The RPV was subsequently depressurized. Repressurization of the RPV has either not occurred (i.e., the containment was vented) or the vessel did repressurize and now is again at low pressure either because a break occurred in the low-pressure ECCS piping or because the containment failed, thus allowing the SRVs to reopen. The condensate system is available. As in Case 1 of this question, all that is required is for the operator to align the system and restart the pumps. The success probabilities are determined in exactly the same manner as for Case 1 above.

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELT:	Impossible	Highly Unlikely	Unlikely
nMELT:	Certain	Highly Likely	Likely

CASE #6:

This case is similar to Case 5 of this question except that HPSW is available for injection rather than the condensate system. Quantification is identical. The discussion under Case 2 is appropriate to this quantification.

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELT:	Impossible	Highly Unlikely	Unlikely
nMELT:	Certain	Highly Likely	Likely

CASE #7:

This case is similar to Case 5 of this question. The vessel pressure is low as described in that case. However, the sequence was initiated by an ATWS (it is TC-core vulnerable), and therefore the operator has far less time than in the TW sequence to restore injection after the containment fails or is vented. This sequence requires only on the order of an hour to reach the point where injection will be lost due to the loss of NPSH for the ECCS pumps. The TW sequence takes about a day and a half. Once injection is lost for the sequence, there are only a few minutes available in which to avert core damage by restoring injection. For the TW sequence, the operators have approximately 2 hours to restore injection after the ECCS is lost. However, this case has condensate available. Since the operators would be very likely to be trying to restore condensate with the ECCS running, quantification of the success probability is the same as for TW. Again, it is judged highly unlikely that the operators would allow the core to melt with the condensate system available.

The optimistic and pessimistic estimates reflect a factor of 100 uncertainty in the failure probability. It is recognized that this quantification is inconsistent with that used by ASEP. The ASEP mean failure probability if close to 0.1 [3].

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELT:	Impossible	Highly Unlikely	Unlikely
nMELT:	Certain	Highly Likely	Likely

CASE #8:

This case is similar to Case 7 of this question with the exception that it is the HPSW on which the operator must rely rather than the condensate system. Because the operator would be less familiar with using the HPSW for injection than with using condensate and because there are more valves that must be opened or closed, it is judged that success is less likely than when condensate is available. Success is judged to be very likely (0.01 probability of failure). The optimistic and pessimistic estimates reflect a factor of 10 uncertainty (0.1 failure probability being the maximum credible).

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELT:	Highly Unlikely	Very Unlikely	Unlikely
nMELT:	Highly Likely	Very Likely	Likely

CASE #9:

This case catches all the ASEP "core melt" sequences. In other words, the previous cases include all sequences identified as being core vulnerable. For this case the core melt probability must thus be unity to maintain consistency with ASEP.

QUANTIFICATION

	Optimistic	Central	Pessimistic
MELT:	Certain	Certain	Certain
nMELT:	Impossible	Impossible	Impossible

QUESTION #36: What is containment pressure before core degradation?

This question provides an assessment of the containment pressure at the onset of core damage. These estimates are based primarily on Source Term Code Package (STCP) [9], MARCON calculations [12], and hand calculations which are documented in Appendix C. The containment pressure depends on both the progression of the accident up until the time core damage begins and the plant damage state, as enumerated by the cases described below. Containment pressures are given in psia.

This is a Type 4 question, and there is only one branch:

1. PBasel: This is not a branch as in the previous questions. This is more of a parameter name and it is the value (in psia) of the containment pressure prior to core damage (pressure due to steam plus heating effects). This "branch" is not referenced below.

CASE #1:

This case covers scenarios in which the initiator is a large LOCA and there is no subsequent water make-up to the vessel. However, containment sprays do operate. (Failure mode for LPCI in injection valve failure to open.) The pressure in containment at the onset of core degradation is low. The BMI-2104 value of 30 psia (Volume II, Table 6.4) is taken as the central estimate. The optimistic value is 17 psia which assumes that containment spray has brought the containment into equilibrium with the pool at 110°F. The pessimistic value is 43 psia which reflects an uncertainty in the central estimate equal to that in the optimistic evaluation.

CASE #2:

This case is similar to Case 1 of this question with the exception that containment spray is not operating. The central estimate of containment pressure is 35 psia. This value is taken from BMI-2104 (Volume II, Table 6.4) and corresponds to the calculated pressure following the blowdown of the RPV. Steam condensation in the pool and on structural heat sinks have lowered the pressure from the peak value of 52 psia (Volume II, Table 6.5) which occurs during the first 30 seconds. The value of 52 psia is used as the pessimistic estimate. the optimistic estimate is 30 psia. This represents the authors' judgment of a reasonable lower bound for this sequence since without drywell spray the drywell will be filled with superheated

190

steam. The lowering from the central estimate reflects uncertainty in the mass and surface area of heat sinks an in the initial pool temperature.

CASE #3:

This case covers small break initiators followed by initial high pressure injection and continued operation of RHR. Injection is lost 2 to 3 hours into the event due to the reduction in RPV pressure. Given the long period of RHR operation an equilibrium condition will have been established in the containment and the pressure will be near atmospheric (pool temperature approaches service water temperature). The central estimate is 17 psia corresponding to an assumption of 2 psia overpressure in the drywell necessary to clear the drywell vents. The optimistic estimate is that the containment pressure is reduced to the ambient value (15 psia) which would occur if the service water temperature were low (60°F). The pessimistic estimate reflects an equal uncertainty in the central evaluation (i.e., 19 psia).

CASE #4:

This case is similar to the previous case with the exception that RHR does not operate initially. The containment pressure is bounded by the results for the AE sequence (Case 2 of this question). The optimistic and pessimistic estimates of containment pressure are the same as for that case. The central estimate is conservative in that the pressure is taken as that resulting from a large break plus 5 psi which is the steam partial pressure increase due to adding 2 hours of decay heat to the pool assuming equilibrium with the containment is achieved. The resultant central estimate is 40 psia (see Appendix C).

CASE #5:

This case considers ATWS sequences in which HPCI fails to operate. Containment sprays are, however, operating. The containment pressure rise will be minimal due to operation of the sprays.

CASE #6:

This case considers transient initiated sequences in which there is no injection to the RPV. Neither sprays nor RHR are available. The pressure rise in containment is that due to the boiloff of the RPV inventory of water alone, the suppression pool is assumed to be at its normal temperature at the start of the accident. The pressure rise in the containment is taken to be equal to the change in the

191

partial pressure of steam which occurs when the pool temperature is raised to 140°F. This is temperature of the pool attained prior to core damage in an STCP calculation of a similar sequence (TBR) for the Grand Gulf plant [10]. No STCP calculation of this sequence has been made for Peach Bottom. However, the ratio of the vessel inventory of water to the mass of the suppression pool is similar between the plants and so the pool temperature after core uncovery is expected to be similar.

CASE #7:

This case considers the long-term station blackout scenario. Injection to the RPV, from HPCI or RCIC, has continued for 6 to 8 hours. Drywell coolers and RHR are unavailable. The drywell temperature is elevated due to heat transfer from the RPV and from the uninsulated discharge tailpipes on the SRVs. This heating of the drywell is anticipated to result in degassing of unlined concrete in the drywell. Both the temperature increase and the liberation of water vapor from the concrete tend to increase containment pressure. In addition, the suppression pool has been heated to approximately 225°F. There is thus expected to be a 19 psi partial pressure of steam in the torus. The CP calculation for this scenario [9] predicts a pressure of 27 psia in the containment at the onset of core damage. Concrete degassing and heat transfer from the RPV to the drywell is not modeled. Thus, the central estimate is 38 psia based on MARCON calculations [11] which model the these phenomena which tend to result in higher pressure predictions.

CASE #8:

This case includes those sequences in which there was a power cycle break. That break could occur at almost any time during the TC sequence, thus the containment pressure at the time of its occurrence is indeterminate. Once the break occurred, the containment pressure would remain essentially constant. Thus, in quantifying the pressure, the approximate range of possibilities is covered by the three walkthroughs. Owing to the low probability of this event the quantification here is expected to have a negligible impact on risk. Optimistically, the containment pressure could be as low as a few psi above normal operating conditions. The central estimate is the mean of the optimistic and pessimistic estimates. Pessimistically, the containment pressure could be up to approximately the threshold for repressurization of the vessel. Once repressurization occurred and low pressure injection ceased there would be little (if any) chance of power cycles.

CASE #9:

This case includes repressurization breaks. Quantification reflects a reasonable variation around the central estimate of the containment pressure at which vessel repressurization occurs. This threshold pressure is determined on the basis of the estimated performance of the SRV accumulators. The containment pressure for a repressurization break would be approximately the pressure at which the accumulator pressure could no longer maintain the valves open. Once the break occurred there would be little further containment pressurizezation. The central estimate thus reflects the best estimate of the containment pressure at which relief mode of the SRVs will fail. The optimistic and pessimistic values reflect a 10% estimated uncertainty in this value.

CASE #10:

In this case the sequence is an ATWS (TC) and adequate core cooling was lost prior to containment failure (i.e., the containment is still intact). Because the containment pressurizes very rapidly in an ATWS there is considerable uncertainty as to the exact pressure. The most likely scenario is that injection is lost at the time the containment pressure reaches the SRV reclosure point. Containment pressure is thus quantified exactly as in the previous case.

CASE #11:

For this case the containment has failed but the leakage rate is insufficient to result in depressurization of the containment. The containment pressure is thus at approximately the ultimate capability of the containment as the central estimate. The optimistic and central estimates represent reasonable variations around the assessed mean containment failure pressure.

CASE #12:

This case catches all sequences in which the containment is either vented or has failed such that the containment is depressurized. With this depressurization the containment pressure is anticipated to be slightly higher than atmospheric. This applies for all walkthroughs. The exact numerical value used here does not impact evaluation of the event tree since the significant fact is that the containment is vented or failed.

QUESTION #37: Will the SP flash following containment vent or rupture?

This question determines whether or not the suppression pool is saturated. (Saturation is defined to mean that the bulk temperature of the pool is equal to the saturation temperature that corresponds to the wetwell airspace pressure.) This is distinct from the pool being in equilibrium with the airspace (partial pressure of steam in the airspace equal to the saturation pressure that corresponds to the pool surface temperature). The significance of the pool being saturated is threefold: First, the pool flashes at containment failure, resulting in the release of large quantities of steam to the reactor building (approximately 360 metric tons if vented at 60 psig, approximately 540 metric tons if failure occurs at 135 psig). Second, in a saturated pool, all the energy deposited in the pool is used to convert water into steam; none of the energy is stored as sensible heat. Third, the ECCS pumps (and RCIC) draw suction from the pool. In BWRs such as Peach Bottom, these pumps cannot pump water that is at or near saturation since the pumps will cavitate. Thus, if the pool saturates on containment failure, there is no ECCS. NOTE: The pool cannot be saturated if the containment is intact but may become so if the containment integrity is not maintained. If the pool is saturated with respect to the ambient pressure (14.7 psia), it is saturated for the purpose of this question.

This is a Type 2 question. and there are two branches:

1. SatSP: The suppression pool will flash and saturate following containment venting or a rupture.
2. SubcSP: The pool temperature is subcooled relative to atmospheric pressure (temperature below 100°C).

CASE #1:

In this case, the high-pressure systems have supplied make-up water to the vessel for sufficiently long that enough heat has been deposited in the pool to raise the temperature to 100°C. The RHR system has not been used to remove the heat. This occurs in TW sequences and in blackouts where there is no battery failure. Under these conditions. the pool will certainly flash when the containment is depressurized.

QUANTIFICATION

	Optimistic	Central	Pessimistic
SatSP:	Certain	Certain	Certain
SubcSP:	Impossible	Impossible	Impossible

CASE #2:

The case considers the possible long-term blackout (TB) scenario in which high-pressure injection fails as a result of elevated suppression pool temperature rather than battery depletion. There is a slim likelihood that this failure occurred at a low enough temperature that the pool is not saturated.

QUANTIFICATION:

	Optimistic	Central	Pessimistic
SatSP:	Likely	Very Likely	Certain
Subc8P:	Unlikely	Very Unlikely	Impossible

CASE #3:

In this case, low-pressure ECCS was operated, but the RHR system was not used to remove the heat deposited in the pool. Since the low-pressure systems did inject, they are certain to run at least until the pool reaches 100°C. This situation arises in all the loss of containment heat removal (W) sequences in which there was no early high-pressure injection (AW, S1W, and TW).

QUANTIFICATION

	Optimistic	Central	Pessimistic
SatSP:	Certain	Certain	Certain
SubcSP:	ImPossible	Impossible	Impossible

CASE #4:

In this case, the sequence is an ATWS in which injection, either high or low-pressure, occurs. Pool heat-up is so rapid that 100°C is reached in no more than 25 minutes. The pool is thus certainly saturated and will flash when containment integrity is lost.

QUANTIFICATION

	Optimistic	Central	Pessimistic
SatSP:	Certain	Certain	Certain
SubcSP:	Impossible	Impossible	Impossible

CASE #5:

This case catches all sequences in which the pool cannot saturate until after vessel breach (if at all). This includes short-term station blackout sequences, AE, S1E, and TQUV.

QUANTIFICATION

	Optimistic	Central	Pessimistic
SatSP:	Impossible	Impossible	Impossible
SubcSP:	Certain	Certain	Certain

QUESTION #38: Is the RPV depressurized during core degradation?

This question determines the RPV pressure at the time of vessel breach.

This is a Type 2 question, and there are two branches:

1. E4nDeP: The RPV is at high pressure (approximately 1100 psig) at vessel breach.
2. E4-DeP: The RPV is depressurized at the time of vessel breach.

CASE #1:

In this case, there has been an induced break in the primary system. Such breaks are assumed always to result in depressurization of the vessel.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nDeP:	Impossible	Impossible	Impossible
E4-DeP:	Certain	Certain	Certain

CASE #2:

In this case, the RPV was previously depressurized, but repressurized during core degradation. Either the containment has failed and dc power is available or ac power has been recovered following a station blackout. Both circumstances will result in the RPV once again depressurizing.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nDeP:	Impossible	Impossible	Impossible
E4-DeP:	Certain	Certain	Certain

CASE #3:

This case includes only the postulated TC sequences in which the operator fails to depressurize the vessel. The ASEP analysis of ATWS provides for this case on the basis that, under stress, the operators may fail to depressurize the vessel once HPCI fails even though the means for low-pressure injection are available. It seems reasonable to allow a significant probability of delayed depressurization for this case. Such depressurization would allow the restoration of injection and the possibility of arresting the accident. The probability of delayed repressurization (i.e., after core damage but prior to vessel breach, a 1-hour time frame according to STCP calculations) is assumed to be 0.6 for the central estimate. While no human factors analysis has been performed, it appears overly conservative to assume that continued failure to depressurize is certain. Depressurization is intrinsic to BWR accident management.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nDeP:	Unlikely	Indeterminate	Likely
E4-DeP:	Likely	Indeterminate	Unlikely

CASE #4:

In this case, power is recovered during core damage that resulted from station blackout. The operators had not previously depressurized the RPV (either because they were unable to or as a result of error). The low-pressure ECCS is available (no failures and suppression pool subcooled). It is all but certain that either ADS or operator action would result in vessel depressurization

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nDeP:	Impossible	Highly Unlikely	Very Unlikely
E4-DeP:	Certain	Highly Likely	Very Likely

CASE #5:

This case is similar to the previous case except that low-pressure ECCS is unavailable either because the suppression pool is saturated (Question #37) or because of failures in the ECCS pumps or valves. The quantification of this case thus reflects the probability that the operators will manually depressurize following ac power recovery given that they failed to depressurize before core damage (in violation of procedure).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nDeP:	Likely	Indeterminate	Certain
E4-DeP:	Unlikely	In8determinate	Impossible

CASE #6:

This case includes sequences in which the vessel has remained at pressure during the entire boil-down phase. Given that depressurization has not occurred prior to core damage it seems reasonable to assume a negligible probability that depressurization will occur during core damage, (unless offsite power is recovered, which is considered in the next case). It should be noted that this case has a negligible frequency of occurrence.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nDeP:	Certain	Certain	Certain
E4-DeP:	Impossible	Impossible	Impossible

CASE #7:

This case catches all sequences in which the vessel has remained depressurized since early in the sequence. Nothing is assumed to change in this respect.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nDeP:	Impossible	Impossible	Impossible
E4-DeP:	Certain	Certain	Certain

QUESTION #39: Is there injection during core degradation?

This question determines whether there was injection of water to the vessel during core degradation. Water injection could result either from the recovery of offsite power (most likely) or from belated operator action (such as vessel depressurization in a TC sequence). Water injection will certainly increase the rate (and probably the amount) of hydrogen production. However, if sufficient water is delivered before the core geometry is badly distorted, vessel breach may be avoided (considered in Question #69). Even if vessel breach is not averted, late injection implies that water will be supplied to core debris in the drywell; this may significantly alter the accident progression and reduce fission product releases.

This is a Type 2 question, and there are three branches:

1. E4-LPIn: Water is delivered to the vessel during core degradation by injection systems that operate only when the vessel pressure is low.

2. E4-CDIn: Water is delivered to the vessel during core degradation only by the CRD hydraulic system.

3. E4nIn: There is no water injection to the vessel during core degradation.

CASE #1:

In this case, at least one of the low-pressure ECC systems is operational and the RPV is at low pressure. Under these conditions, water injection is automatic and is therefore judged to be certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-LPIn:	Certain	Certain	Certain
E4-CDIn:	Impossible	Impossible	Impossible
E4nIn:	Impossible	Impossible	Impossible

CASE #2:

Scenarios in which ECC was available were considered by the previous case. Thus, for this case, LPCS and LPCI are unavailable. However, the RPV is depressurized and the condensate system is available. The quantification of branch probabilities below reflects the authors' assessment that the operators will align the condensate system for injection prior to RPV meltthrough.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-LPIn:	Certain	Highly Likely	Very Likely
E4-CDIn:	Impossible	Impossible	Impossible
E4nIn:	Impossible	Highly Unlikely	Very Unlikely

CASE #3:

This case is similar to the previous one. However, condensate in addition to ECC, is failed. HPSW is available. The probability that injection will be restored is lower for this case since the operators are less familiar with the use of this system for injection to the RPV.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-LPIn:	Highly Likely	Likely	Indeterminate
E4-CDIn:	Impossible	Impossible	Impossible
E4nIn:	Highly Unlikely	Unlikely	Indeterminate

CASE #4:

In this case the CRD pumps have been providing injection. They are certain to continue to do so. (Possible induced failures have previously been considered.)

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-LPIn:	Impossible	Impossible	Impossible
E4-CDIn:	Certain	Certain	Certain
E4nIn:	Impossible	Impossible	Impossible

CASE #5:

This case covers sequences in which no injection system is available. There is thus certainly no injection to the vessel.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-LPIn:	Impossible	Impossible	Impossible
E4-CDIn:	Impossible	Impossible	Impossible
E4nIn:	Certain	Certain	Certain

QUESTION #40: What is the status of containment sprays?

This question determines the availability of the drywell sprays. Drywell spray is a mode of the RHR. In spray mode, the RHR pumps draw suction from the suppression pool and return the water through the RHR heat exchangers (cooled by emergency service water, if available) to spray headers in the drywell at the approximate elevation of the vessel head. Because spray water can significantly impact fission product transport, with a dependence on spray water temperature, which is small compared to other uncertainties in fission product transport calculations, the presence or absence of the heat removal function of the spray is not considered here but is left to be included in questions regarding the status of RHR. Thus, if both spray and RHR were shown to be in use (for example), then this could mean that the operator was using spray mode for containment heat removal. (It is also possible that one RHR loop is in spray mode and the other is in pool-cooling mode. However, for the purposes of the event tree the difference is unimportant.) The use of HPSW for containment spray is not considered at this stage in the accident. Such use is considered at late times (Question #87). During the core damage phase, it is presumed that the operators will focus on restoring RPV injection. Extraordinary actions directed at mitigating symptoms of core damage (e.g., high containment pressure and temperature) are extremely unlikely.

This is a Type 2 question, and there are three branches:

1. E4fCS: Containment spray mode of the RHR system is failed.

2. E4rCS: Containment spray mode of the RHR system is unavailable but may be recovered.

3. E4-CS: Containment sprays are actuated.

CASE #1:

In this case, containment sprays have failed previously. They thus remain failed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fCS:	Certain	Certain	Certain
E4rCS:	Impossible	Impossible	Impossible
E4-CS:	Impossible	Impossible	Impossible

CASE #2:

In this case containment sprays would operate if ac power were available but there has been no recovery of ac power. Thus, the sprays remain in the recoverable condition.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fCS:	Impossible	Impossible	Impossible
E4rCS:	Certain	Certain	Certain
E4-CS:	Impossible	Impossible	Impossible

201

CASE #3:

In this case, the RHR pumps have been available to operate the sprays. Since core damage is occurring, the containment pressure and/or temperature will be above the technical-specification limit. Thus the operator would be expected to use sprays to reduce temperature and/or pressure. While the Emergency Operating Procedures would indicate the use of sprays, this is a manual action, and the operators may be so concerned with the vessel status that containment spray will not be actuated. Quantification of the spray probability thus reflects the possibility that the operators concentrate on restoring vessel injection. It is the authors' judgment that the probability of failure to use sprays is negligible since the use of spray is a normal procedure for the operators to use in addressing accident situations.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fCS:	Impossible	Impossible	Impossible
E4rCS:	Impossible	Impossible	Impossible
E4-CS:	Certain	Certain	Certain

CASE #4:

In this case, the capability for spray has been recovered as a consequence of restoration of offsite power. An increased probability for operator failure to activate sprays is applied for this case, relative to the previous one. It is judged that there may be considerable confusion when power is restored (with many systems returning to operation) which will lead to increased chance for error.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fCS:	Impossible	Unlikely	Indeterminate
E4rCS:	Impossible	Impossible	Impossible
E4-CS:	Certain	Likely	Indeterminate

CASE #5:

This case includes sequences in which there are no water sources to operate the sprays or the RHR pumps are failed. The sprays are thus failed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4fCS:	Certain	Certain	Certain
E4rCS:	Impossible	Impossible	Impossible
E4-CS:	Impossible	Impossible	Impossible

QUESTION #41: What is the level of gas flow to the drywell during CD?

This question determines the extent to which the suppression pool is bypassed during core degradation. Bypass in this sense refers to gas (and steam) flow from the RPV which passes to the drywell before having the opportunity to pass through the suppression pool. Two levels of bypass are considered. Level 2 bypass is associated with a partial bypass with the majority of the flow being from the RPV directly to the suppression pool. Level 3 bypass indicates that essentially all the flow from the RPV passes directly to the drywell. Flow going to the drywell will bypass the suppression pool completely if the drywell is failed. If the drywell is intact, the effectiveness of the suppression pool in scrubbing radionuclides from the releases is reduced for this release pathway.

This is a Type 2 question, and there are three branches:

1. E4nFDW: All flow from the RPV prior to core damage goes through the SRVs to the suppression pool.
2. E4-FDW2: A substantial fraction of the flow from the RPV goes directly to the drywell rather than through the SRV quenchers in the suppression pool.
3. E4-FDW3: All flow from the RPV during core damage goes directly to the drywell.

CASE #1:

In this case, there is either a moderate-to-large break (S1 or A) in the nuclear steam supply system, or there is a small break (S2/3) and the RPV has been depressurized. In either case, essentially all the effluent from the RPV is assumed to pass directly into the drywell.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nFDW:	Impossible	Impossible	Impossible
E4-FDW2:	Impossible	Impossible	Impossible
E4-FDW3:	Certain	Certain	Certain

CASE #2:

In this case, there is an induced break in the low pressure ECCS piping. In the CET, this case is treated as flow from the RPV directly to the drywell and then from the drywell to the reactor building. While this case does not appear in the current results, if a source term calculation were to be made, deposition in the drywell would be set to zero.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nFDW:	Impossible	Impossible	Impossible
E4-FDW2:	Impossible	Impossible	Impossible
E4-FDW3:	Certain	Certain	Certain

CASE #3:

This case considers a small-break LOCA in which the RPV remains pressurized. The flow split between the drywell and the suppression pool is approximately equal to the ratio of the effective area of the break to the effective area of one SRV. A significant portion of the flow will go to the drywell; but in the size range considered as a small break, the majority will go to the suppression pool.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nFDW:	Impossible	Impossible	Impossible
E4-FDW2:	Certain	Certain	Certain
E4-FDW3:	Impossible	Impossible	Impossible

CASE #4:

In this case, there is a stuck open SRV tailpipe vacuum breaker. The RPV is pressurized. With the RPV pressurized, flow-through holes in the SRV tailpipe are in proportion to the flow area (choked flow). Stuck open tailpipe vacuum breakers have small flow areas compared to the T-quencher (< 15%). Thus, the majority of the flow will pass through the pool. Level 2 bypass, as defined above, is thus the worst outcome. The opinions of the SARRP expert reviewers were used in the quantification of this case. Level 2 bypass corresponds to essentially the entire area of the vacuum breaker being available for flow. If the vacuum breaker merely leaks, this flow is neglected in the analysis.

204

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nFDW:	Very Likely	Likely	Impossible
E4-FDW2:	Very Unlikely	Unlikely	Certain
E4-FDW3:	Impossible	Impossible	Impossible

CASE #5:

This case is similar to the previous one, with the exception that the vessel pressure is low. The flow split between the drywell and the T-quenchers is no longer in proportion to the flow area since the quenchers are submerged in the suppression pool and thus have the 5 psi head of water relative to the drywell. If the tailpipe vacuum breaker is stuck slightly open, there will still be only a Level 2 bypass. The consensus opinion of the experts who reviewed the Peach Bottom risk issues for SARRP [6] was that the vacuum breaker was more likely to leak (80% chance given failure to reclose). Thus, the probability of complete bypass is taken as 0.2 (central estimate). This is applied to the small break case on the basis that it is a reasonable value. The pessimistic evaluation considers complete bypass to be certain. Optimistically, a 0.01 probability of complete bypass is assigned.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nFDW:	Impossible	Impossible	Impossible
E4-FDW2:	Very Likely	Likely	Impossible
E4-FDW3:	Very Unlikely	Unlikely	Certain

CASE #6:

This case covers accident progressions in which the RPV pressure boundary is intact and the SRV tailpipe vacuum breakers reseat correctly. All effluents from the RPV pass to the suppression pool.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nFDW:	Certain	Certain	Certain
E4-FDW2:	Impossible	Impossible	Impossible
E4-FDW3:	Impossible	Impossible	Impossible

QUESTION #42: What is the H2 generation and pressure rise during CD?

This question determines the amount of hydrogen generated during core degradation. Three possible levels of hydrogen production are considered. The question quantifies the level of hydrogen production in terms of the pressure rise (in psi) in the primary containment that results from hydrogen generation, steam production and temperature increase. In each case, the optimistic walkthrough corresponds to low hydrogen production (Branch 1), the central walkthrough to medium hydrogen production (Branch 2), and the pessimistic walkthrough to high hydrogen production (Branch 3). Three sources are used to assess the pressure rise values: (1) STCP calculations [9], (2) MARCON calculations [12], and (3) available MAAP calculations [13]. It is emphasized that the pressure rise here is not that due to hydrogen alone, but rather the total pressure increase in containment that occurs during the in-vessel hydrogen production phase of the accident.

This is a Type 4 question, and there are three branches:

1. H2Low: Hydrogen production in-vessel is in the range predicted by IDCOR's MAAP code for BWRs (4 to 10% of the available zirconium).

2. H2Med: Hydrogen production in-vessel is in the range predicted by MARCH (i.e., 30 to 60% of the active fuel clad).

3. H2Hi: Hydrogen production in-vessel is in the range predicted by MARCON (i.e., equivalent to 40 to 80% of the available zirconium).

CASE #1:

In this case, hydrogen production, regardless of magnitude, cannot pressurize the primary containment because its integrity has been previously violated. The pressure rise is thus zero for all walkthroughs.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Certain	Impossible	Impossible
H2Med:	Impossible	Certain	Impossible
H2Hi:	Impossible	Impossible	Certain

206

CASE #2:

In this case, injection has been restored during core damage. In addition, flow from the RPV goes to the drywell rather than the suppression pool (determined by Question #41). Depending on the timing of injection being restored, significantly larger amounts of hydrogen may be produced. This possibility is reflected by assigning a 50-50 split, in the central estimate, between what may be considered a central value for pressure rise and the pessimistic value. The splits between high, medium, and low hydrogen production used for this case reflect engineering judgment only. No calculations have been made, that the author is aware of, which would substantiate these values. The central estimate pressure rise is based on the STCP calculation of the TB1 scenario [9] and accounts for additional hydrogen production and heating of the drywell due to hot gases that are not cooled by the suppression pool. A correction for low RPV pressure is necessary since the TB1 calculation has the RPV at high pressure. In TB1, the RPV contains a large quantity of hydrogen immediately before RPV meltthrough. Had the RPV been depressurized earlier in the accident, it is assumed that this same quantity of hydrogen (less some small portion which would be in the depressurized RPV) would be present in the containment airspace.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Indeterminate	Impossible	Impossible
H2Med:	Indeterminate	Indeterminate	Impossible
H2Hi:	Impossible	Indeterminate	Certain

CASE #3:

As in the previous case, this case considers accident scenarios in which injection to the RPV is restored during core damage. This case is distinguished from Case 2 in that there is little (if any) direct flow from the RPV to the drywell. Drywell temperatures are thus lower, as is the anticipated containment pressure increase which results from core damage. Again, pressure rise values used for this case reflect engineering judgment based on available (but not really applicable) STCP calculations.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Indeterminate	Impossible	Impossible
H2Med:	Indeterminate	Indeterminate	Impossible
H2Hi:	Impossible	Indeterminate	Certain

CASE #4:

This case considers a large- or intermediate-break LOCA. Containment pressure rise during core damage is taken from the BMI-2104 [18] calculation of the AE sequence. The large pressure spike associated with core slumping in that calculation is not included. This was done since it is the analyst's opinion that the slump scenario is unlikely. The pessimistic estimate includes this pressure spike.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Certain	Impossible	Impossible
H2Med:	Impossible	Certain	Impossible
H2Hi:	Impossible	Impossible	Certain

CASE #5:

In this case, flow from the CRD pumps continues throughout core damage. Containment pressure rise was derived based on the hydrogen production calculated for the TC1 scenario by the STCP [9]. Pressure was estimated assuming the containment was intact and that temperatures were those calculated for the TC2 scenario (which does not include CRD flow). RPV pressure may be either high or low for this case. A distinction was not made since the steam generated from the CRD flow would prevent large quantities of hydrogen accumulating in the vessel as is observed in the TB1 scenario. Thus, for either high or low pressure in the RPV, most of the hydrogen (which would account for most of the pressure increase) is in the containment.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Certain	Impossible	Impossible
H2Med:	Impossible	Certain	Impossible
H2Hi:	Impossible	Impossible	Certain

CASE #6:

This case considers transients or small-break sequences in which essentially all flow from the RPV passes into the drywell (determined from Question #41). STCP results for the TB1 scenario [9] were used as the basis for the pressure rise estimates. A correction was made to account for the elevated drywell temperature which is caused by direct leakage from the RPV. The pressure rise here is thus appropriate for a scenario in which the RPV is at high pressure. No distinction based on RPV pressure is made. In actuality, most scenarios considered under this case would have the RPV at low pressure. The pressure rise assigned for this case may therefore be lower than is appropriate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Certain	Impossible	Impossible
H2Med:	Impossible	Certain	Impossible
H2Hi:	Impossible	Impossible	Certain

CASE #7:

In this case, the RPV was depressurized during core damage. Scenarios which evolve from the TCSX and TCUX plant damage states are the most likely considerations for this case, although other damage states are possible. The pressure rise assumed is the same as that assumed for scenarios in which the RPV is always depressurized. While blowdown of the RPV during core damage may increase hydrogen production, no calculations were available which could be used to quantify this effect. A separate case is maintained to ensure that the potential difference is recognized. The pressure rise value for this case was estimated based on the STCP TB1 results [9]. That scenario assumed a high pressure RPV. A correction was made to adjust for the fact that the RPV was at low pressure by adding the partial pressure which would be exerted on the containment by the hydrogen which was in the high pressure RPV immediately before vessel breach. The gas temperature assumed in the partial pressure calculation was that calculated for the drywell in the TB1 immediately prior to RPV meltthrough.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Certain	Impossible	Impossible
H2Med:	Impossible	Likely	Impossible
H2Hi:	Impossible	Unlikely	Certain

CASE #8:

This case covers the TB plant damage state for scenarios in which there has been no power recovery. This case is separated from other accident progression in which the RPV is pressurized simply because an STCP calculation is available which is directly applicable (TB1). The central estimate pressure rise was obtained from that calculation.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Certain	Impossible	Impossible
H2Med:	Impossible	Certain	Impossible
H2Hi:	Impossible	Impossible	Certain

CASE #9:

This case considers accident progressions in which core damage occurs with the RPV at high pressure. Those stemming from the TB plant damage state are excluded by the previous case. Thus, this case covers short-term scenarios (e.g., TBUX, TCSX, and TCUX). A slightly higher pressure rise is used than is used for the TB case. This is to account for concrete degassing, which occurs before core damage in TB (and is accounted for in Question #36), but will occur to a substantial degree during core damage for short-term scenarios.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Certain	Impossible	Impossible
H2Med:	Impossible	Certain	Impossible
H2Hi:	Impossible	Impossible	Certain

CASE #10:

This case covers accident progressions in which the RPV is at low pressure throughout core damage. The basis for the pressure rise estimate is described under Case 8 for this question.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2Low:	Certain	Impossible	Impossible
H2Med:	Impossible	Certain	Impossible
H2Hi:	Impossible	Impossible	Certain

QUESTION #43: Does at least one drywell vacuum breaker stick open?

This question determines whether a drywell vacuum breaker (wetwell to drywell) sticks open during the in-vessel phase of the accident. These vacuum breakers are essentially check valves that open to prevent suppression pool water being pushed back through the drywell vent pipes should the wetwell pressure exceed that in the drywell. Being check valves, there is some probability that they will stick in the open position. If the vacuum breakers are open when there is flow from the drywell to the wetwell, the suppression pool is at least partially bypassed. The extent of bypass depends on the position in which the vacuum breaker sticks and is considered in Question #58 below. In quantifying the probability that a vacuum breaker will stick in the open position, the WASH 1400 probability of failure per demand was used. The median value in WASH-1400 is 1.1 E-4. The mean value is 3.3 E-4. This value is assumed for the first demand. The failure probability is assumed to be a factor of 10 less on subsequent demands, provided the RPV is at low pressure. With the RPV at high pressure, no reduction is assumed since more than one vacuum breaker would be challenged and any given device may open for the first time. The number of demands on the vacuum breakers was estimated assuming a vacuum breaker opens once for every psi the containment pressure rises above atmospheric. Pressure rise was evaluated using the values given in Question #36 and Question #42 above (calculations are shown in Appendix C).

This is a Type 2 question, and there are two branches:

1. E4-VBo: At least one of the drywell vacuum breakers is stuck in the open position.
2. E4nVBo: None of the drywell vacuum breakers is stuck open.

CASE #1:

In this case, there is a primary system break and all flow from the primary system is to the drywell. The drywell pressure is thus higher than the wetwell, and the vacuum breakers do not open and cannot stick open (prior to containment failure). This case also includes sequences in which containment failure occurred in the wetwell prior to core degradation. If this is the case, pressurization of the wetwell relative to the drywell cannot be established and, again, the vacuum breakers are not opened.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-VBo:	Impossible	Impossible	Impossible
E4nVBo:	Certain	Certain	Certain

CASE #2:

In this case, the sequence is either a TC (ATWS) or a TW. In both these sequences, the containment pressure rises to (or beyond) the failure pressure. Over one hundred challenges to the vacuum breaker(s) are to be expected. Quantification assumes 110 demands in the central estimate. For the pessimistic case the failure probability is increased by a factor of 10. The optimistic case is the same as the central to allow for the possibility that a breaker sticks open (the failure rate per demand is believed to be an optimistic evaluation but is the only one available to use as the central evaluation).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-VBo:	Highly Unlikely	Highly Unlikely	Very Unlikely
E4nVBo:	Highly Likely	Highly Likely	Very Likely

CASE #3:

In this case, the sequence is either transient-initiated and an SRV tailpipe vacuum breaker is stuck open or the sequence was initiated by a small primary system break. For either case, the RPV pressure is low. With the vessel pressure low, a significant fraction of the flow from the vessel will pass into the drywell. This will reduce (perhaps to near zero) the number of challenges to the drywell vacuum breakers. Since failure probabilities are low, a crude estimate is made for this case by taking 50% of the failure probability in Case 2 of this question. (There is an inaccuracy in the specification of this case which could be removed had reference been made to Question #41. The current logic here assumes SRV tail pipe vacuum breakers always stick wide open. As a result, there is a slight underestimate of the suppression pool bypass probability.)

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-VBo:	Impossible	Highly Unlikely	Very Unlikely
E4nVBo:	Certain	Highly Likely	Very Likely

CASE #4:

In this case, the vessel is at low pressure and all the flow from the vessel is to the wetwell. Optimistically, there is a 20 psi increase in containment pressure (20 challenges to the vacuum breakers). The central estimate is 46. The pessimistic value is 113. Failure probabilities are reduced for demands after the first as described above.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-VBo:	Highly Unlikely	Highly Unlikely	Very Unlikely
E4nVBo:	Highly Likely	Highly Likely	Very Likely

CASE #5:

In this case, the sequence is transient-initiated, vessel pressure is high, and the amount of in-vessel hydrogen production is assumed to be high (see description under Question #42). The optimistic estimate is that there are 20 challenges. The central estimate is 40. The pessimistic value is 97.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-VBo:	Very Unlikely	Very Unlikely	Unlikely
E4nVBo:	Very Likely	Very Likely	Likely

CASE #6:

This case is similar to the previous case (Case 5), with the exception that hydrogen production is moderate rather than high. Since high hydrogen production is approximately twice the moderate (central) value, the failure probability for this case is taken to be 50% of that in the previous case (i.e., the number of demands is assumed to be reduced by a factor of 2).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-VBo:	Impossible	Very Unlikely	Unlikely
E4nVBo:	Certain	Very Likely	Likely

CASE #7:

This case covers transient sequences in which the RPV is at high pressure and hydrogen production is low. Low production is approximately 20% of the central estimate, thus the vacuum breaker failure (to reclose) probability is taken as 20% of that for the previous case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-VBo:	Impossible	Highly Unlikely	Very Unlikely
E4nVBo:	Certain	Highly Likely	Very Likely

QUESTION #44: Does H2 discharge lead to SP bypass?

This question considers the possibility of suppression pool bypass, resulting from structural damage to the drywell vent system and induced by the release of hydrogen from the vessel under high pressure. When core degradation occurs with RPV at high pressure, the low-low set SRV opens periodically in relief mode. Once the valve opens, pressure within the RPV drops approximately 50 psi before the valve will reclose. Assuming that the volume available to gas (steam plus hydrogen) within the vessel is 12,000 cubic feet and that the gas mixture behaves as ideal, if the mean RPV pressure is 1070 psia and the mean gas temperature is 2000 R, then each valve opening (pop) releases nearly 11,000 SCF of gas to the pool. A 50% volume fraction of hydrogen is not unlikely (10% by mass). Since SRV is open for less than 2.5 seconds (0.1 ft opening, 2100 ft/s choked velocity), even if all the steam were condensed, a 5000 cubic foot bubble could be formed at the quencher (equivalent of a 10-ft radius). This bubble has the potential to do over 350 thousand foot-pounds of work (0.5 megajoules), the energy of a small car traveling 10 mph. Clearly, this magnitude of energy release has the potential for damaging the wetwell structure. Both the vent pipes and vent header could be damaged. Either would result in a bypass of the suppression pool. A bounding estimate of the loading on a downcomer has been made. The resultant load was found to to be approximately 25% of the design capability of the downcomers. It is thus considered that the probability of bypass from this mechanism is negligible. This question, and the one which follows, are left in the tree for the sake of completeness. Consideration of these two questions is appropriate if the CET is modified for application to a plant other than Peach Bottom.

This is a Type 2 question, and there are two branches:

1. H2-SPB: Structural damage to the drywell vent system (from hydrogen discharge under high pressure) has resulted in suppression pool bypass.

2. H2nSPB: No structural damage, resulting in pool bypass, has occurred in the wetwell.

CASE #1:

In this case, the RPV is at low pressure and thus the potential for structural damage via the mechanism described above does not exist. This case includes sequences initiated by primary system breaks.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2-SPB:	Impossible	Impossible	Impossible
H2nSPB:	Certain	Certain	Certain

CASE #2:

This case (and those following) covers those sequences during which there is discharge of hydrogen to the suppression pool under high pressure. In this case, hydrogen production is high, and thus discharge to the suppression pool has the highest volume fraction of hydrogen. Since the in-vessel hydrogen concentration is estimated to be on the order of 95% (see analysis for Question #42 in Appendix C) the energy released with each bubble is approximately twice that for the central estimate (50% concentration in-vessel). The probability for bypass is still assumed negligible since the load would remain below the design value.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2-SPB:	Impossible	Impossible	Impossible
H2nSPB:	Certain	Certain	Certain

CASE #3:

This case is similar to the previous case (2) except that the hydrogen production rate in-vessel is assumed to be moderate. As stated above, the probability of induced bypass is negligible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2-SPB:	Impossible	Impossible	Impossible
H2nSPB:	Certain	Certain	Certain

CASE #4:

This case covers sequences in which hydrogen production is low and the vessel is at high pressure. The in-vessel hydrogen concentration is on the order of 10%. The bypass probabilities are thus clearly negligible also.

QUANTIFICATION

	Optimistic	Central	Pessimistic
H2-SPB:	Impossible	Impossible	Impossible
H2nSPB:	Certain	Certain	Certain

QUESTION #45: Does H2 discharge to SP induce WW failure?

This question considers the possibility that the torus is damaged by the discharge of hydrogen to the suppression pool under high pressure (described in Question #44), resulting in leakage from the containment. If the torus were damaged, at best the containment could be vented from the wetwell airspace; at worst, the suppression pool could be drained. Since our best engineering judgment is that there is a negligible probability of bypass due to structural damage from hydrogen release, negligible probability for containment failure is implied by this previous answer to the question. Two cases of high pressure discharge are considered. The first is the case in which suppression-pool bypass has been induced (previous question). For this case the central estimate of the probability that the torus is also failed is 0.1. This value is selected simply to include the possibility. The second case is that in which no damage to the drywell vent system was induced. Since water elevated by pool swell would strike the vent system with greater force than it would the torus wall, induced failure of the torus without pool bypass is considered essentially incredible.

This is a Type 2 question, and there are three branches:

1. E4nHWF: There is no structural damage to the wetwell resulting from the release of hydrogen under high pressure.

216

BEST COPY AVAILABLE

2. E4-HWF2: The release of hydrogen from the vessel results in a Level 2 leak in the torus. The leakage rate is insufficient to result in significant depressurization of the containment.

3. E4-HWF3: There is structural damage to the torus as a result of high pressure hydrogen release. A rupture of the torus has resulted and the containment depressurizes.

CASE #1:

In this case, the RPV was depressurized early in the sequence and did not repressurize prior to core damage. Core degradation thus occurs at low pressure and there is no potential for structural damage in the torus due to hydrogen release.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nHWF:	Certain	Certain	Certain
E4-HWF2:	Impossible	Impossible	Impossible
E4-HWF3:	Impossible	Impossible	Impossible

CASE #2:

In this case, there has been structural damage to the vent-header system. The probability that the torus is also damaged is assumed to be 0.1. This case does not apply to the current tree but is included for completeness.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nHWF:	Very Likely	Likely	Indeterminate
E4-HWF2:	Impossible	Impossible	Impossible
E4-HWF3:	Very Unlikely	Unlikely	Indeterminate

CASE #3:

This case covers sequences in which hydrogen discharge did not cause damage to the vent-header system sufficient to result in suppression pool bypass. The probability of structural damage to the torus is zero.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nHWF:	Certain	Certain	Certain
E4-HWF2:	Impossible	Impossible	Impossible
E4-HWF3:	Impossible	Impossible	Impossible

QUESTION #46: Is the vent threshold reached during core degradation?

This question instructs the event-tree code to calculate the sum of the containment pressure at the onset of core damage (Parameter #1) and the pressure rise due to in-vessel hydrogen production (Parameter #2). This sum is compared to the vent threshold pressure of 75 psia (60 psig) stated in the Peach Bottom Emergency Operation Procedures. If the sum is greater that 60 psig, Branch 1 is taken. If not, Branch 2 is taken.

This is a Type 6 question, and there are two branches:

1. E4-Vth: The containment pressure does exceed the vent threshold pressure during core degradation.
2. E4nVth: The containment pressure does not exceed the vent threshold pressure of 60 psig during core degradation.

CASE #1:

This case includes all sequences in which there has been no failure of the wetwell (torus) induced by the discharge of hydrogen from the vessel under high pressure. Containment failure modes other than this (described in Question #45) are accounted for in the base pressure used in the evaluation of the pressure at the end of core degradation (Pbasel defined in Question #36 above). EVNTRE thus evaluates whether the vent threshold has been reached for the remaining sequences in which the containment is intact during core degradation.

CASE #2:

This case includes all sequences in which there has been a rupture of the wetwell (torus) as a result of hydrogen being discharged from the vessel under high pressure. The containment is thus failed, and the vent threshold will never be reached. A dummy value of the vent threshold (999 psi rather than 75) is used as the parameter in this question to insure this case always results in Branch 2 (vent threshold not reached) being taken. This case does not apply to the current tree since the probability of torus failure due to hydrogen discharge is zero.

QUESTION #47: Does containment venting not occur during core degradation?

This question determines whether or not the containment is vented during core degradation (prior to vessel meltthrough if core damage is not arrested in-vessel). Venting is instructed by the emergency procedures once the containment reaches 60 psig. There is no qualification of this procedure to account for radiation levels in the containment. Venting would be carried out as described under Question #23 above.

This is a Type 2 question, and there are two branches:

1. E4nVENT: The containment is not vented during core degradation (i.e., prior to vessel meltthrough).
2. E4 VENT: The containment is vented during core degradation; prior to vessel meltthrough.

CASE #1:

In this case, the containment was vented prior to core damage but the size of the vent area was inadequate to reduce containment pressure. This situation occurs only for ATWS sequences. By the time core damage begins, steam (and hydrogen) generation within the containment has been reduced to a level that will allow the containment to depressurize. Thus, what was unsuccessful venting becomes successful.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nVENT:	Impossible	Impossible	Impossible
E4-VENT:	Certain	Certain	Certain

CASE #2:

In this case, the sequence is a TW. The venting threshold was exceeded long before core damage occurred. If the operator did not vent before, now that there is high radiation in the containment, the probability that he will vent now is judged to be negligible.

219

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nVENT:	Certain	Certain	Certain
E4-VENT:	Impossible	Impossible	Impossible

CASE #3:

In this case, the containment developed a leak (insufficient to cause depressurization) prior to core damage. ATWS is the most likely sequence considered, although sequences involving loss of containment heat removal would be treated. The quantification used here is appropriate for ATWS. The containment has not been vented, which, given current procedures, constitutes an error. However, containment pressurization occurred on a very short time scale. It is thus possible that core damage occurred before operators could open the vent. An indeterminate probability for belated success is used in the central estimate. No human reliability analysis has been performed. However, the analysts judge that there would be a significant probability of venting.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nVENT:	Very Unlikely	Indeterminate	Impossible
E4-VENT:	Very Likely	Indeterminate	Certain

CASE #4:

This case considers all nonblackout scenarios in which the containment integrity is preserved but the pressure has exceeded a threshold for venting (determined by the previous question, Question #46). The most likely scenarios here are the TCSX and TCUX damage states. These are similar to the TC2 scenario which was analyzed with the STCP [9]. Idaho National Engineering Laboratory (INEL) has assessed the probability of venting for the TC2 scenario [5]. This assessment was used as the basis for quantification of this case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nVENT:	Very Unlikely	Unlikely	Certain
E4-VENT:	Very Likely	Likely	Impossible

CASE #5:

For both this case and the one succeeding it, core damage was initiated by station blackout. Offsite power is recovered during core damage. The containment pressure exceeds the threshold for venting. This case considers situations in which containment pressure cannot be reduced below the venting threshold other than by venting. Specifically, either RHR or containment spray has failed, or else hydrogen production accounts for most of the pressure (high generation rate determined by Question #42). The probability of failure to vent for this case is based on an assessment by the ASE analysts.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nVENT:	Highly Unlikely	Very Unlikely	Unlikely
E4-VENT:	Highly Likely	Very Likely	Likely

CASE #6:

The conditions for this case are similar to those for the previous case. However, RHR and/or drywell spray are expected to significantly reduce pressure in the containment. Venting is judged to be unlikely if the operators believe they can control containment pressure by more traditional means.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nVENT:	Certain	Likely	Indeterminate
E4-VENT:	Impossible	Unlikely	Indeterminate

CASE #7:

This case is a station blackout, and there is no power available to permit venting of the containment by remote actuation of the valves from the control room. Based on the EOPs, the operators would be instructed to open the 6-inch ILRT line. Opening this line would require personnel to go into the reactor building to the valves located on the floor above the torus room. These valves could be opened manually, and it is judged that until the last of the four valves was opened, radiation levels would be tolerable. However, as the last valve was opened, highly radioactive gas would pass through this pipe adjacent to where the personnel were standing. It seems likely that the personnel involved would at least be seriously injured.

Even though this case includes only sequences in which the vent threshold pressure has been exceeded, it is far from certain that the operators would be aware of this fact. As discussed under Question #23 above, containment pressure indication requires ac power. Thus, for this case, venting is considered unlikely. Quantification of this case agrees with the ASEP estimate. The mean failure probability is 0.9, with a maximum (pessimistic) value of 1.0 and a minimum (optimistic) value of 0.5.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nVENT:	Indeterminate	Likely	Certain
E4-VENT:	Indeterminate	Unlikely	Impossible

CASE #8:

This case catches all sequences for which the venting threshhold is not reached during core degradation. If the threshold is not reached, the EOPs do not instruct venting. Thus the probability of venting is assumed to be negligible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nVENT:	Certain	Certain	Certain
E4-VENT:	Impossible	Impossible	Impossible

QUESTION #48: Does the RPV repressurize during core degradation?

This question is analogous to Question #24 above. However, in this question EVNTRE is instructed to calculate whether the containment pressure exceeds the level above which the SRV accumulators cannot hold the valves open. The containment pressure at the onset of core damage (Parameter #1) is summed with the pressure rise due to hydrogen generation (Parameter #2). If the sum exceeds the threshold value, Branch 1 is taken and repressurization is assumed to occur.

This is a Type 6 question, and there are two branches:

1. E4-ReP: The RPV repressurizes as a result of containment pressurization during core degradation.
2. E4nReP: The RPV does not repressurize during core degradation.

CASE #1:

In this case, the initiating event is a transient, there are no stuck open relief valves, and the containment has not failed or been vented. Thus repressurization is possible, and the code proceeds with the calculation described above.

CASE #2:

This case is the complement of Case 1 and includes all sequences in which the RPV could not repressurize. In this case, Parameter #2 is excluded from the sum, which assures that repressurization will not be indicated.

QUESTION #49: Is there a repressurization break during core degradation?

This question is analogous to Question #26 above. Refer to the description there.

This is a Type 2 question, and there are two branches:

1. E4-RePB: There is a break in one of the low pressure systems as a result of RPV repressurization during core degradation.
2. E4nRePB: There is no break in the low pressure systems as a result of RPV repressurization during core degradation.

CASE #1:

In this case, the vessel repressurizes during core degradation. Quantification of the probability of repressurization is identical to Question #26 above (i.e., there is a negligible probability of an induced LOCA).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-RePB:	Impossible	Impossible	Impossible
E4nRePB:	Certain	Certain	Certain

CASE #2:

This case covers all sequences in which there is no repressurization of the RPV and thus a repressurization break is impossible.

223

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-RePB:	Impossible	Impossible	Impossible
E4nRePB:	Certain	Certain	Certain

QUESTION #50: Does the containment fail by pressure during core degradation?

This question instructs the code to calculate the probability of containment failure during core degradation. The containment fragility is assumed to be normally distributed. The mean failure pressure is estimated based on optimistic, central, and pessimistic assumptions. The code calculates the sum of the containment pressure at the onset of core degradation (Parameter #1) and the pressure rise due to hydrogen generation (Parameter #2). Based on the fragility, the probability of containment failure (of Branch 1 being taken) is then calculated. The central estimate of the containment failure probability is based on an assumed mean failure pressure of 132 psia. This value was calculated by Ames Laboratory on the basis of a 2% strain criteria for yielding (failure) [7]. It is the authors' opinion that this value is conservative. However, for the purposes of SARRP, we are using evaluations which can be referenced in the literature as far as possible in the central walkthrough. A 5-psi standard deviation has been assumed in the central estimate to reduce the probability of containment failure significantly below 132 psia to a minimum. Uncertainty in the failure pressure is considered in the limited Latin Hypercube uncertainty study. In that study, the predominance of the uncertainty is judged to be on the side of higher failure pressures.

This is a Type 6 question, and there are two branches:

1. E4-CFP: The containment fails during core degradation.
2. E4nCFP: There is no failure of the containment during core degradation.

CASE #1:

In this case, the containment has not previously failed or been vented and there was no repressurization break resulting in containment bypass during core degradation. Failure due to overpressure is a possibility, and the probability is calculated as described above.

CASE #2:

In this case, the containment has been vented or has previously failed. The case also includes sequences in which there has been a repressurization break. The code will assign zero probability for overpressure failure in this case.

QUESTION #51: What is location of containment leakage prior to vessel breach?

This question determines the location of containment leakage (if any) prior to vessel failure. The question is analogous to Question #28 and in the case that containment failure is predicted to occur during core degradation, it is quantified in the same manner. It has generally been assumed that once the containment develops leakage in a certain location, all further leakage will be at the same location. Exceptions apply in the case of containment venting.

This is a Type 2 question, and there are three branches:

1. E4nCL: There is no leakage from within the containment during core damage in excess of technical specifications.
2. E4-CLD: The leakage from within the containment during core damage comes from the drywell (or from within the RPV).
3. E4-CLW: The leakage from within the containment during core damage comes from the wetwell.

CASE #1:

In this case, either there was previously leakage from the drywell or a repressurization break (induced LOCA) occurred during core degradation (Question #49). Previous leakage may also include induced primary system breaks. Since leakage from the drywell has been previously established, it is judged certain to continue to be the dominant mode. Even if the previous failure were a leak, subsequent overpressure failure (since the leak was too small to relieve the pressure), it is assumed certain to be at the same location. Implicit in the latter assumption is the belief that a leak indicates structural weakness at the location of the leak and a predilection for exacerbated leakage or rupture to occur at the same point.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CLD:	Certain	Certain	Certain
E4-CLW:	Impossible	Impossible	Impossible

CASE #2:

In this case, leakage from the wetwell (torus) has previously been established. This leakage could result from (1) previous overpressurization, (2) leakage induced as a result of hydrogen discharge to the suppression pool, or (3) containment venting. Following the logic delineated in the previous case, further leakage is judged certain to be from the wetwell.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CLD:	Impossible	Impossible	Impossible
E4-CLW:	Certain	Certain	Certain

CASE #3:

In this case, there has been an overpressure failure of the containment during core degradation (Question #50). The quantification here, and its basis, is identical to Case 4 of Question #28.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CLD:	Indeterminate	Likely	Certain
E4-CLW:	Indeterminate	Unlikely	Impossible

CASE #4:

This case catches all sequences in which there has been no breach of the containment. Thus, leakage is from neither the wetwell or drywell, and "no significant leakage" is certain for all walkthroughs.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Certain	Certain	Certain
E4-CLD:	Impossible	Impossible	Impossible
E4-CLW:	Impossible	Impossible	Impossible

QUESTION #52: For drywell failures, is the failure at the head seal?

This question determines whether the location of leakage from the drywell is from the head seal, given that the failure location is in the drywell. This question is completely analogous to Question #29.

This is a Type 2 question, and there are three branches:

1. E4-DHF: Containment leakage during core damage is from the drywell head, and thus, flow is directly from the drywell to the refueling bay.

2. E4-DKF: Containment leakage during core damage is from the drywell to the reactor building' (characteristics are similar to leakage from the knuckle).

3. E4nDF: The drywell remains intact during core damage (no leakage from the drywell).

CASE #1:

Leakage was previously from the drywell head. The containment has not been vented. Leakage is assumed certain to continue to be from the head region.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-DHF:	Certain	Certain	Certain
E4-DKF:	Impossible	Impossible	Impossible
E4nDF:	Impossible	Impossible	Impossible

CASE #2:

Leakage was previously from the drywell to the reactor building. Either the earlier failure was a rupture (meaning that even if venting occurred it would be ineffectual) or the containment was not vented. This logic excludes early leakage-type failures which are subsequently vented (Question 47, Case 3). It is assumed certain that leakage will continue to be from the same location in the drywell.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-DHF:	Impossible	Impossible	Impossible
E4-DKF:	Certain	Certain	Certain
E4nDF:	Impossible	Impossible	Impossible

CASE #3:

Containment overpressure failure occurs in the drywell during core damage (based on branch taken in Question #51). This case is quantified exactly the same as Case 3 of Question #29.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-DHF:	Unlikely	Indeterminate	Likely
E4-DKF:	Likely	Indeterminate	Unlikely
E4nDF:	Impossible	Impossible	Impossible

CASE #4:

In this case, either the drywell remains intact or venting subsequent to earlier leakage from the head seal has resulted in the torus becoming the dominant location for leakage. There is no leakage from either the drywell head seal or the knuckle.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4-DHF:	Impossible	Impossible	Impossible
E4-DKF:	Impossible	Impossible	Impossible
E4nDF:	Certain	Certain	Certain

QUESTION #53: What is the level of containment leakage before VB?

This question determines the level (or rate) of leakage from the containment at the time of vessel breach. A full description of the failure level definitions is given in Question #30 above.

This is a Type 2 question, and there are three branches:

228

1. E4nCL: There is no leakage from the containment significantly in excess of technical specifications prior to vessel breach.

2. E4-CL2: There is leakage from the containment that is significantly in excess of technical specifications prior to vessel breach but insufficient to cause the containment to depressurize.

3. E4-CL3: There is a containment breach prior to containment failure that is sufficiently large to cause the containment to depressurize.

CASE #1:

In this case, the containment previously ruptured; leakage will certainly persist at the same level. This case includes accident progressions resulting from the TCSRX2 plant damage state in which leakage was induced in the containment at locations other than the head seal. The effective area of these leaks would be large enough that they should be classified as ruptures (i.e., the containment will depressurize) once the steam generation which resulted from the ATWS has decreased. The head closure is assumed to reseat, reducing the effective leakage area and, thus, this initial leakage location for TCSRX2 is excluded from this case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CL2:	Impossible	Impossible	Impossible
E4-CL3:	Certain	Certain	Certain

CASE #2:

In this case, the containment is completely bypassed through a primary system break in the reactor building. The terminology here is somewhat confusing since depressurization of (leakage from) the containment would not begin until vessel breach. However, the main purpose of this question is to determine the impact of leakage rate on the reactor-building thermal-hydraulics and on fission-product transport. For sequences with containment bypass, containment behavior prior to vessel breach does not have a significant impact on the consequence evaluation so this slight ambiguity is acceptable. Subsequent questions properly distinguish containment-bypass sequences (which are expected to have negligible impact on risk due to their low frequency) from sequences with a rupture in the containment. A break in low pressure systems is assumed to be large (pump-seal failures are regarded to be negligible leaks), this case is thus assumed to be certain to have Level 3 leakage.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CL2:	Impossible	Impossible	Impossible
E4-CL3:	Certain	Certain	Certain

CASE #3:

In this case the containment has been successfully vented. Successful venting has been defined to mean that the containment depressurizes. Thus this case is certain to result in Level 3 leakage in all walkthroughs.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CL2:	Impossible	Impossible	Impossible
E4-CL3:	Certain	Certain	Certain

CASE #4:

In this case, the containment failed as a result of overpressurization. The failure location was either in the wetwell or in the drywell at a location other than the head seal (see Question #29). The evaluation of this case is identical to Case 6 of Question #30.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CL2:	Likely	Unlikely	Impossible
E4-CL3:	Unlikely	Likely	Certain

CASE #5:

In this case, containment leakage due to overpressure occurs at the drywell head seal. It was the consensus of the experts consulted (Case 7 of Question #30) that this failure mode would always be a leak type of failure (Level 2).

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CL2:	Certain	Certain	Certain
E4-CL3:	Impossible	Impossible	Impossible

CASE #6:

This case covers all sequences in which Level 2 leakage has previously been established. No further challenge to the containment integrity has occurred so the leakage is assumed certain to persist at the same level.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Impossible	Impossible	Impossible
E4-CL2:	Certain	Certain	Certain
E4-CL3:	Impossible	Impossible	Impossible

CASE #7:

This case covers sequences in which there has been no challenge to the containment sufficient to result in leakage. "No significant leakage" is thus certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E4nCL:	Certain	Certain	Certain
E4-CL2:	Impossible	Impossible	Impossible
E4-CL3:	Impossible	Impossible	Impossible

QUESTION #54: What is the status of CRD?

As noted above (Question #33), there is the possibility that steam flooding of the reactor building resulting from containment failure could impact the operability of the CRD pumps for the majority of Mark I and II BWRs. In this question, the possibility of hydrogen burns in the reactor building impacting CRD pump operability must also be considered since hydrogen would be released from the containment subsequent to core damage if its integrity was violated. Since in the Peach Bottom plant, the CRD pumps are located in the turbine building, there is assumed to be a negligible probability of induced failure.

This is a Type 2 question, and there are three branches:

1. E5fCRD: The CRD pumps are not (or are no longer) available to supply make-up water to the vessel.
2. E5rCRD: The CRD pumps are unavailable but may be recovered.
3. E5-CRD: The CRD pumps are injecting water to the vessel.

CASE #1:

In this case, the containment pressure has been relieved to the reactor building (rupture or vent). Large quantities of steam, and perhaps large quantities of hydrogen, would be released into it. The CRD pumps are operating at the time of the failure. As stated above, the probability of failure being induced is considered negligible. Quantification is identical to Case 1 of Question 33.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fCRD:	Impossible	Impossible	Impossible
E5rCRD:	Impossible	Impossible	Impossible
E5-CRD:	Certain	Certain	Certain

CASE #2:

This case is like the previous one except that CRD pumps are unavailable due to the unavailability of ac power. Their recoverability may be impacted by the containment leakage that has occurred. Quantification of the failure probability is identical to the previous case. This is based on the assumption that the potential for damage is independent of whether the system is energized at the time containment integrity is lost.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fCRD:	Impossible	Impossible	Impossible
E5rCRD:	Certain	Certain	Certain
E5-CRD:	Impossible	Impossible	Impossible

CASE #3:

In this case, there was previously flow from the CRD system to the vessel. The containment either has not ruptured, ruptured prior to core degradation (with no effect on the CRD pumps), or was vented to the atmosphere via the ILRT line. This case also includes sequences in which there is only Level 2 (defined in Question #30) leakage from the containment to the reactor building. Even if the CRD pumps were in the reactor building, there would be little potential for steam damage in this case. The probability that the CRD pumps would fail is thus zero.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fCRD:	Impossible	Impossible	Impossible
E5rCRD:	Impossible	Impossible	Impossible
E5-CRD:	Certain	Certain	Certain

CASE #4:

This case is similar to the previous one except that it is the recoverability that is being examined rather than the operating status.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fCRD:	Impossible	Impossible	Impossible
E5rCRD:	Certain	Certain	Certain
E5-CRD:	Impossible	Impossible	Impossible

CASE #5:

This case covers sequences in which the CRD pumps previously failed. They are certain to remain so.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fCRD:	Certain	Certain	Certain
E5rCRD:	Impossible	Impossible	Impossible
E5-CRD:	Impossible	Impossible	Impossible

QUESTION #55: What is the status of HPSW?

This question treats the status of HPSW much as the question above treated the status of CRD flow. There is a significant difference. As with the CRD pumps, the HPSW pumps will not be impacted by steam flooding of the reactor building or by hydrogen burns. As was noted in Question #34 (above), the HPSW pumps are outside the the reactor building, and only valve operability could be impacted. Thus, if HPSW were already aligned to the LPCI system, there would be a negligible probability of any impact. If HPSW were not in service, conditions in the reactor building could impact its operability. Note that in the branch definitions, failure is taken to include failure of the operator to use the system.

This is a Type 2 question, and there are four branches:

1. E5fHPSW: The High-Pressure Service Water system cannot be (or is not due to operator error) aligned to provide water to the LPCI system.
2. E5rHPSW: The High-Pressure Service Water system remains unavailable but may be recovered.
3. E4aHPSW: The High-Pressure Service Water System is available for supplying water to the LPCI system.
4. E5-HPSW: The High-Pressure Service Water system is supplying water to the LPCI system.

CASE #1:

In this case, injection to the RPV was reestablished during core damage. Both low-pressure ECC and the condensate pumps are failed. It is thus implied that injection is accomplished with HPSW.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fHPSW:	Impossible	Impossible	Impossible
E5rHPSW:	Impossible	Impossible	Impossible
E4aHPSW:	Impossible	Impossible	Impossible
E5-HPSW:	Certain	Certain	Certain

CASE #2:

This case considers station blackout scenarios. The HPSW system remains unavailable but recoverable, implying that offsite power has not been restored. The containment has been vented. Venting without ac power is accomplished using the ILRT line. This line leads directly from the torus to the outside wall of the reactor building. Since no steam will enter the reactor building from this line, HPSW remains recoverable.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fHPSW:	Impossible	Impossible	Impossible
E5rHPSW:	Certain	Certain	Certain
E4aHPSW:	Impossible	Impossible	Impossible
E5-HPSW:	Impossible	Impossible	Impossible

CASE #3:

This case considers station blackout scenarios in which HPSW becomes available during core damage. Recovery of offsite power is implicit in that situation. The containment was vented prior to core damage (without benefit of ac power). As for the previous case, the reactor building is bypassed by the line used for venting. HPSW is thus certain to remain available.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fHPSW:	Impossible	Impossible	Impossible
E5rHPSW:	Impossible	Impossible	Impossible
E4aHPSW:	Certain	Certain	Certain
E5-HPSW:	Impossible	Impossible	Impossible

CASE #4:

In this case, High-Pressure Service Water is available, but the operator has not previously aligned it to the LPCI system. There is steam flooding and hydrogen in the reactor building (cause as in Case #1 of Question #54). Two factors are considered in the quantification. First the probability that the reactor building environment will cause one or more of the required valves to fail in the closed position. This is quantified exactly as in the subsequent case of this question with regard to failure of the HPSW system, given that it was previously recoverable. The second factor is the probability that the operator will take the actions required to align the system. In the central estimate (and the optimistic), the probability of failure for each cause is judged equal (unlikely). The quantification for these two walkthroughs is thus twice that in Case #1 of Question #34. For the pessimistic estimate, a failure probability larger than 50% is hard to justify; thus, this is the estimated failure probability.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fHPSW:	Highly Unlikely	Unlikely	Indeterminate
E5rHPSW:	Impossible	Impossible	Impossible
E4aHPSW:	Highly Likely	Likely	Indeterminate
E5-HPSW:	Impossible	Impossible	Impossible

CASE #5:

In this case, the containment depressurized into the reactor building and the HPSW system was previously in an unavailable but recoverable condition. Quantification of this question reflects the probability that the resultant severe reactor-building conditions will cause at least one of the motor operated valves required to align HPSW to the LPCI system to fail in the closed position. Identical to Case #2 of Question #34.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fHPSW:	Highly Unlikely	Unlikely	Indeterminate
E5rHPSW:	Highly Likely	Likely	Indeterminate
E4aHPSW:	Impossible	Impossible	Impossible
E5-HPSW:	Impossible	Impossible	Impossible

CASE #6:

In this case, there has been no rupture of containment, and the HPSW system was previously available. It is assumed that any leakage from containment would not impact the valves' operability. HPSW is thus certain to remain available.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fHPSW:	Impossible	Impossible	Impossible
E5rHPSW:	Impossible	Impossible	Impossible
E4aHPSW:	Certain	Certain	Certain
E5-HPSW:	Impossible	Impossible	Impossible

CASE #7:

In this case, HPSW was unavailable but recoverable at the onset of core damage. The containment remains intact. There is no possibility of damage to the HPSW valves. HPSW remains recoverable.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fHPSW:	Impossible	Impossible	Impossible
E5rHPSW:	Certain	Certain	Certain
E4aHPSW:	Impossible	Impossible	Impossible
E5-HPSW:	Impossible	Impossible	Impossible

CASE #8:

The case catches all sequences in which the HPSW system was previously failed. It is thus certain to remain so.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fHPSW:	Certain	Certain	Certain
E5rHPSW:	Impossible	Impossible	Impossible
E4aHPSW:	Impossible	Impossible	Impossible
E5-HPSW:	Impossible	Impossible	Impossible

QUESTION #56: Is MSIV leakage control or condenser vacuum maintained?

This question determines the availability of the MSIV Leakage Control System (MSIV-LCS). The purpose of the MSIV-LCS is to direct fission products, which may leak from the MSIVs after they have closed, to the SGTS system. If condenser vacuum is maintained, the same end is achieved. The MSIV-LCS consists of two (inboard and outboard) safety systems. They are powered by the emergency busses. Actuation is by manual action from the control room. Interlocks prevent actuation if the main steam line pressure is high (not a concern here since MSIVs are assumed closed). This manual actuation is a normal part of the post-LOCA procedure. This question is not referenced in the current tree but is included since this may be of interest in source-term binning. For Peach Bottom, possible influence on risk is probably minor, owing to the large releases associated with the dominant core melt sequences.

This is a Type 2 question, and there are two branches:

1. MSIVnLC: There is no MSIV leakage control.

2. MSIV-LC: The MSIV leakage control system is operating.

CASE #1:

In this case, ac power is available. The operation of the MSIV Leakage Control System depends only on action by the operator to start the system. There is some possibility that this would not be done and that the condenser vacuum pumps were not running. It is noted that actuation of this system is a routine part of operator response to a LOCA or other accident resulting in MSIV closure. The central estimate is thus that the probability of failure is highly unlikely (0.001). The optimistic and pessimistic estimates reflect a factor of 100 uncertainty (large enough to have noticeable impact in the pessimistic walkthrough).

QUANTIFICATION

	Optimistic	Central	Pessimistic
MSIVnLC:	Impossible	Highly Unlikely	Unlikely
MSIV-LC:	Certain	Highly Likely	Likely

CASE #2:

This case covers sequences in which there is no ac power. It is certain that the MSIV Leakage Control System would be inoperable and that there would be no condenser vacuum.

QUANTIFICATION

	Optimistic	Central	Pessimistic
MSIVnLC:	Certain	Certain	Certain
MSIV-LC:	Impossible	Impossible	Impossible

QUESTION #57: Is the suppression pool drained before VB?

This question determines the availability of water in the torus to provide for pressure suppression, pool scrubbing, and low pressure ECC. Given that one allows for wetwell failures, one must consider failures that may occur below the pool surface. Failure occurring more than about 2 meters below the water level would drain the pool sufficiently that flow coming through the drywell vents would not be impacted by the pool (i.e., flow through the drywell vents would completely bypass the pool). This question is analogous to Question #31 above and is used to determine the status of the suppression pool immediately prior to vessel breach.

This is a Type 2 question, and there are three branches:

1. E5-SPD: The suppression pool has been essentially drained as a result of failure below the water level combined with a failure of the torus room to contain the water escaping the pool.

2. E5pSPD: The suppression pool level has decreased, as a result of a failure in the torus, such that the downcomers from the drywell vent ring header are essentially uncovered.

3. E5nSPD: The suppression pool maintains at least normal operating level.

CASE #1:

In this case, the suppression pool was drained prior to core degradation. It is assumed certain that it is drained at vessel breach.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-SPD:	Certain	Certain	Certain
E5pSPD:	Impossible	Impossible	Impossible
E5nSPD:	Impossible	Impossible	Impossible

CASE #2:

This case considers accident progressions in which the pool was partially drained prior to core damage. Since it was assumed that the torus room retained the water leaked from the pool, the suppression-pool level is assumed to remain essentially constant at the reduced level. Bypass by flows through the drywell vents (only) is assured.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-SPD:	Impossible	Impossible	Impossible
E5pSPD:	Certain	Certain	Certain
E5nSPD:	Impossible	Impossible	Impossible

CASE #3:

In this case, the torus ruptures during core damage. Quantification of this case is identical to Case #1 of Question 31.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-SPD:	Impossible	Very Unlikely	Unlikely
E5pSPD:	Very Unlikely	Indeterminate	Likely
E5nSPD:	Very Likely	Indeterminate	Unlikely

CASE #4:

In this case, the release of hydrogen to the pool under high pressure has resulted in a large breach in the torus (Question #45). In the central estimate, it is judged that this failure is likely to be above the pool level. This is because the hydrogen bubbles would tend to lift the pool, throwing water against the upper part of the torus. The fall back under gravity would be far less energetic. The energy released in this case is discussed under Question #44 above.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-SPD:	Impossible	Very Unlikely	Very Unlikely
E5pSPD:	Very Unlikely	Unlikely	Indeterminate
E5nSPD:	Very Likely	Likely	Indeterminate

CASE #5:

In this case, there is no large failure of the torus. Leaks are assumed to be of insufficient size to result in a significant loss of pool inventory. Thus the pool is certain to maintain an adequate inventory of water.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-SPD:	Impossible	Impossible	Impossible
E5pSPD:	Impossible	Impossible	Impossible
E5nSPD:	Certain	Certain	Certain

QUESTION #58: What is the level of suppression-pool bypass at VB?

This question determines the extent to which flows from the primary system into the containment or the reactor building (or to the environment) bypass the suppression pool. Pool bypass determines the pressure suppression ability of the containment as well as the fission product removal (due to pool scrubbing) along the release pathway. Some additional explanation of the branch definitions is required. For the purposes of this question, a Level 2 bypass is taken to mean that the pressure-suppression function of the suppression pool is still available but fission products that are gradually released to the drywell airspace will most likely bypass the pool (i.e., the DF for vent flows is 2 or less due to bypass). This would occur with a Level 2 leak in the drywell. Level 3 bypass indicates complete bypass of the suppression pool by all released fission products and gases.

This is a Type 2 question, and there are three branches:

1. E5nSPB: The suppression pool is not bypassed. Gas flow from the vessel to the containment is directed toward the pool.
2. E5-SPB2: There is some significant amount of pool bypass. More than 50% of the flow leaving the vessel will bypass the pool.
3. E5-SPB3: The suppression pool is effectively completely bypassed. It is assumed that no pressure suppression or fission product scrubbing can occur.

CASE #1:

In this case, either the drywell is ruptured or the suppression pool has drained down to the level at which the SRV quenchers are uncovered. In either case, complete bypass of the suppression pool is assured.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nSPB:	Impossible	Impossible	Impossible
E5-SPB2:	Impossible	Impossible	Impossible
E5-SPB3:	Certain	Certain	Certain

CASE #2:

In this case, the containment is completely bypassed since there is a primary system break outside containment. The pool is also certainly bypassed.

241

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nSPB:	Impossible	Impossible	Impossible
E5-SPB2:	Impossible	Impossible	Impossible
E5-SPB3:	Certain	Certain	Certain

CASE #3:

In this case, either a drywell vacuum breaker has stuck open or the discharge of hydrogen from the RPV when it was at high pressure (Question #44) has resulted in damage to the drywell vents (ring header or downcomers) sufficient to cause pool bypass. For either failure mechanism, suppression pool bypass is assumed to be complete.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nSPB:	Impossible	Impossible	Impossible
E5-SPB2:	Impossible	Impossible	Impossible
E5-SPB3:	Certain	Certain	Certain

CASE #4:

In this case there is Level 2 leakage from the drywell, but the suppression pool is intact. This case defines Level 2 bypass (which is therefore certain in this case). With slow pressurization of the drywell, the majority of the flow will escape containment through the drywell leak. With rapid pressurization, most of the flow will go through the vents.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nSPB:	Impossible	Impossible	Impossible
E5-SPB2:	Certain	Certain	Certain
E5-SPB3:	Impossible	Impossible	Impossible

CASE #5:

This case covers all sequences in which the drywell, its vent system, and the suppression pool are intact. There is thus no pool bypass.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nSPB:	Certain	Certain	Certain
E5-SPB2:	Impossible	Impossible	Impossible
E5-SPB3:	Impossible	Impossible	Impossible

QUESTION #59: What is level of RB breach/bypass before VB without burn?

This question determines the extent to which fission products could be retained in the reactor building (assuming they are released from the primary containment). There is an implicit assumption that fission product retention varies inversely with the extent to which the reactor building is bypassed or with the flow rate of aerosol through it. This assumption may be in conflict with the IDCOR model, which appears to rely heavily on aerosol impaction for fission product removal. This question does not consider the impact of possible hydrogen burns in the reactor building. In order to correctly represent the sequence progression, the impact of hydrogen burns (if any) are determined in Question #63, below.

This is a Type 2 question, and there are four branches:

1. E5nRBB: There is no breach or bypass of the reactor building. What flow may exist to the reactor building could be handled by the Standby Gas Treatment System (SGTS) if it is operational.

2. E5-RBB2: There is leakage from the reactor building, but the leakage flow from the primary containment passes through the building.

3. E5-RBB3: The reactor building is essentially bypassed. Either primary containment leakage is directly to the refueling bay, or there is sufficient damage to the reactor building that retention along the release pathway would be similar to that case.

4. E5-RBB4: The reactor building is completely bypassed. There would be no significant fission product retention.

CASE #1:

In this case, there is a primary system break in the reactor building. During core degradation leakage flow to the reactor building is small (compared to that resulting from a rupture in the containment) but large enough to cause the reactor building blowout panels to lift. However, the low-pressure ECCS piping (assumed to be the release pathway) is low in the reactor building. Thus the released gasses must pass completely through the building in order to be released. The bypass level is thus certain to be Level 2.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nRBB:	Impossible	Impossible	Impossible
E5-RBB2:	Certain	Certain	Certain
E5-RBB3:	Impossible	Impossible	Impossible
E5-RBB4:	Impossible	Impossible	Impossible

CASE #2:

In this case, the sequence is a station blackout. Venting occurred without ac power. The vent line that must have been used is the 6-inch ILRT line (described under Question #24). This line is made of piping and passes through the reactor building, at the next level above the torus, exhausting outside the building. With this line open, the reactor building is bypassed completely.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nRBB:	Impossible	Impossible	Impossible
E5-RBB2:	Impossible	Impossible	Impossible
E5-RBB3:	Impossible	Impossible	Impossible
E5-RBB4:	Certain	Certain	Certain

CASE #3:

In this case, there has been an overpressure failure of the containment at the drywell head seal. The containment leakage level for this failure mode is always assumed to be Level 2 (Question #53). Leakage from the head seal passes around the refueling plug and into the refueling bay. This constitutes a Level 3 bypass by definition. Thus, the probability of Level 3 bypass is unity for all walkthroughs.

It is noted that there may be substantial retention of aerosols on the pathway around the refueling plug. Consideration of this is left to the source term calculation.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nRBB:	Impossible	Impossible	Impossible
E5-RBB2:	Impossible	Impossible	Impossible
E5-RBB3:	Certain	Certain	Certain
E5-RBB4:	Impossible	Impossible	Impossible

CASE #4:

In this case, the containment is successfully vented. It is assumed that the 6-inch ILRT line that would be used in a blackout (see previous Case #2) would not be employed for nonblackout sequences. Thus the reactor building would not be completely bypassed as in that case. However, opening one or both of the 18-inch vent lines would cause rapid pressurization of the reactor building and loss of its integrity. A partial bypass (Level 2) is thus certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nRBB:	Impossible	Impossible	Impossible
E5-RBB2:	Certain	Certain	Certain
E5-RBB3:	Impossible	Impossible	Impossible
E5-RBB4:	Impossible	Impossible	Impossible

CASE #5:

In this case, there has been a rupture of the containment as a result of overpressurization. Ruptures are assumed to occur either in the drywell, at or below the knuckle, or in the torus. Either location of primary containment failure results in a leakage pathway to the reactor building. This situation corresponds to the definition of Level 2 reactor building bypass.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nRBB:	Impossible	Impossible	Impossible
E5-RBB2:	Certain	Certain	Certain
E5-RBB3:	Impossible	Impossible	Impossible
E5-RBB4:	Impossible	Impossible	Impossible

CASE #6:

In this case, there is leakage from the containment, but the leakage rate is low compared to that associated with a rupture or venting. There is significant probability that the Standby Gas Treatment System (SGTS) could handle such flow rates. Structural heat sinks may absorb enough heat from gases leaving the primary containment to prevent the hot gas from closing the fire dampers in the SGTS system. Quantification thus allows for the possibility of no breach or bypass. The central estimate for the probability of partial (Level 2) bypass is 0.5. This reflects the fact that the quantification is based solely on the authors' engineering judgment. The optimistic and pessimistic evaluations reflect what is judged to be a reasonable range of uncertainty. (An input error resulted in the pessimistic estimate being incorrectly quantified.)

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nRBB:	Certain	Indeterminate	Likely
E5-RBB2:	Impossible	Indeterminate	Unlikely
E5-RBB3:	Impossible	Impossible	Impossible
E5-RBB4:	Impossible	Impossible	Impossible

CASE #7:

This case catches sequences for which there is no significant leakage from the primary containment prior to vessel breach. This being the case, the reactor building remains intact. If ac power is available, SGTS will be running.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5nRBB:	Certain	Certain	Certain
E5-RBB2:	Impossible	Impossible	Impossible
E5-RBB3:	Impossible	Impossible	Impossible
E5-RBB4:	Impossible	Impossible	Impossible

QUESTION #60: Are the fire sprays actuated before VB without burn?

This question determines whether the fire sprays in the reactor building are triggered. Triggering due to hydrogen burns is considered in a subsequent question. This question considers only the impact of hot gases being released to the building prior to vessel breach. Fire sprays will start automatically when the building temperature is raised significantly above normal operating limits.

The sprays in Peach Bottom are limited to a curtain spray that is designed only to prevent the spread of fire from the reactor building around one unit to that portion surrounding the other unit. Thus the Peach Bottom fire sprays are considered to have negligible impact on either accident progression or fission product releases. This is certainly not the case for all Mark I containments. The fire sprays are independent of plant ac power since the pumps are driven by dedicated fire system diesels.

This is a Type 2 question, and there are two branches:

1. EFireS: Fire sprays are actuated in the reactor building prior to vessel breach.

2. nEFireS: Fire sprays are not actuated in the reactor building as a result of steam being released from the containment prior to vessel breach.

CASE #1:

In this case, there is significant leakage from the containment to the reactor building. (Sequences with leakage directly to the refueling bay are excluded since such leakage is not expected to result in fire spray actuation.) The reactor building is not bypassed, and so the gases released from primary containment are anticipated to accumulate. The suppression pool is either bypassed or saturated such that the released gases are certain to be sufficiently hot that the fire sprays will be actuated. Thus, in this case, fire sprays are certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
EFireS:	Certain	Certain	Certain
nEFireS:	Impossible	Impossible	Impossible

CASE #2:

This case is the complement of case one. Sequences with the following characteristics are included: (1) the containment is intact and no significant quantity of gas is released to the reactor building, (2) the reactor building is bypassed so there is no accumulation of released gases within the building, and (3) the suppression pool is not bypassed and is subcooled. In the latter instance any gases released to the reactor building will be relatively cool. If any of the above three conditions are met, the reactor building will not heat up, and the fire sprays will not be triggered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
EFireS:	Impossible	Impossible	Impossible
nEFireS:	Certain	Certain	Certain

QUESTION #61: Does SGTS fail before VB without burn?

This question determines the availability of the Standby Gas Treatment System (SGTS). The SGTS is a safety system designed to limit the leakage of particulate fission products from the reactor building. When activated, SGTS draws air (or whatever gas is present) from the reactor building and exhausts it through charcoal filters up the stack. At Peach Bottom, the SGTS lines contain fire dampers that are intended to close if there is a fire in the reactor building. Dampers are secured open by fusible links. These links fail at approximately 160°F causing the dampers to close (by the action of gravity). Thus, when the reactor building temperature is raised, the fire dampers close and SGTS fails. The SGTS draws power from the emergency busses and will not operate in a station blackout.

This is a Type 2 question, and there are three branches:

1. E5fSGTS: The Standby Gas Treatment System is failed.

2. E5rSGTS: The Standby Gas Treatment System is unavailable but may be recovered.

3. E5-SGTS: The Standby Gas Treatment System is operating.

CASE #1:

In this case, the reactor building temperature has not been raised sufficiently high to actuate the fire sprays (discussed in the previous question). The containment is not vented (which may cause SGTS to isolate--see Case #2 of this question). Lack of fire sprays implies that the SGTS dampers will be open (high temperatures in the reactor building would both start the fire sprays and close the fire dampers). Only sequences with ac power are included in this case so all the conditions are met for SGTS to be operating. It is thus certain to be operating.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fSGTS:	Impossible	Impossible	Impossible
E5rSGTS:	Impossible	Impossible	Impossible
E5-SGTS:	Certain	Certain	Certain

CASE #2:

In this case, the primary containment has been vented to SGTS. Ac power is available but the reactor building fire sprays are not operating. Since venting is directly to SGTS, the fire dampers may close even though steam leaks to the reactor building are not large enough to result in fire spray operation. Thus, there is judged to be a significant probability of SGTS failure.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fSGTS:	Unlikely	Indeterminate	Likely
E5rSGTS:	Impossible	Impossible	Impossible
E5-SGTS:	Likely	Indeterminate	Unlikely

CASE #3:

This case includes station blackouts in which the reactor-building temperature is still low enough not to result in actuation of the fire systems. The SGTS thus is unavailable due to the lack of ac power. However, the dampers remain open so the system could be recovered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fSGTS:	Impossible	Impossible	Impossible
E5rSGTS:	Certain	Certain	Certain
E5-SGTS:	Impossible	Impossible	Impossible

CASE #4:

This case covers all sequences in which the fire sprays start due to high reactor building temperature. It is thus expected that the fire dampers will close, failing SGTS.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5fSGTS:	Certain	Certain	Certain
E5rSGTS:	Impossible	Impossible	Impossible
E5-SGTS:	Impossible	Impossible	Impossible

QUESTION #62: Does hydrogen burn in RB before VB?

This question determines whether or not there are hydrogen burns in the reactor building. This becomes possible if there is a loss of containment integrity and the reactor core is damaged. Even though the primary containment is inerted, the reactor building contains sufficient air that even if it is mixed with all the nitrogen escaping from the primary containment, hydrogen combustion could still be supported. Hydrogen burning is assumed to occur even in the absence of electrical power since ignition is very easily achieved. However, it is also assumed that if large quantities of steam are released to the reactor building, the building will initially be steam-inerted. Numerous calculations, using HECTR, MELCOR, and CONTAIN [14, 15] have been performed to assess the impact of hydrogen burns on the performance of the reactor building a Peach Bottom. For hydrogen sources that result from STCP calculations, combustion in the reactor building is virtually assured. This conclusion is very different from that reached by IDCOR. Combustion of flammable gases in the reactor building was found not to be of concern in the IDCOR work.

This is a Type 2 question, and there are two branches:

1. HBbVB: Hydrogen burns in the reactor building prior to vessel breach.
2. nHBbVB: There is no hydrogen burn in the reactor building prior to vessel breach.

CASE #1:

In this case, there is a primary system break in the reactor building. Very hot hydrogen and superheated steam will be released directly into the air of the reactor building. Combustion is judged to be certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
HBbVB:	Certain	Certain	Certain
nHBbVB:	Impossible	Impossible	Impossible

CASE #2:

In this case, the reactor building has been completely bypassed. With the reactor building totally bypassed in this manner, no significant amount of hydrogen can accumulate in it, and thus, there will be no combustion.

QUANTIFICATION

	Optimistic	Central	Pessimistic
HBbVB:	Impossible	Impossible	Impossible
nHBbVB:	Certain	Certain	Certain

CASE #3:

In this case, hydrogen production has been assumed to be low (on the order of that predicted by the MAAP code for BWRs). With only a few hundred pounds released from the in-vessel reaction of Zircaloy, combustible concentrations will not accumulate in the reactor building. Thus the probability of a significant hydrogen burn is judged to be negligible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
HBbVB:	Impossible	Impossible	Impossible
nHBbVB:	Certain	Certain	Certain

CASE #4:

In this case, there is some leakage of steam, hydrogen, and nitrogen from the primary containment. However, the leakage is not sufficient to depressurize the primary containment (Level 2). The SGTS is operating and thus it seems unlikely that combustible hydrogen concentrations would accumulate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
HBbVB:	Impossible	Unlikely	Indeterminate
nHBbVB:	Certain	Likely	Indeterminate

CASE #5:

In this case, the reactor building is substantially bypassed (leakage is essentially directly to the refueling bay). The SGTS is not operating. Hydrogen accumulation within the refueling bay is expected. It is thus assumed that hydrogen combustion would be highly likely in the central estimate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
HBbVB:	Indeterminate	Highly Likely	Certain
nHBbVB:	Indeterminate	Highly Unlikely	Impossible

CASE #6:

In this case, there is a large leak from the primary containment into the reactor building. The Standby Gas Treatment System is operating. It is judged highly unlikely that SGTS will prevent combustible hydrogen concentrations from accumulating in the reactor building, given the large leakage level from the containment.

QUANTIFICATION

	Optimistic	Central	Pessimistic
HBbVB:	Indeterminate	Highly Likely	Certain
nHBbVB:	Indeterminate	Highly Unlikely	Impossible

CASE #7:

This case covers sequences in which there is leakage from the containment but no SGTS operation. Calculations have been performed at SNL and ORNL, using various codes (CONTAIN, HECTRE, and MELCOR), that demonstrate that burns will occur in the reactor building if there is a large leak from the containment and an ignition source is present (these calculations considered core-concrete interaction hydrogen sources, but the initial burn occurs before a significant amount from this source has left the primary containment). The presence of an ignition source is assumed. If ac or dc power is available, there is little question of this assumption being valid. If no power is available, an ignition source such as hot equipment or sparks from metal objects moved by the flow of gases has been assumed. For low leakage rates, in the absence of a large bypass of the reactor building and without SGTS operation, combustible concentrations are also certain to accumulate eventually.

QUANTIFICATION

	Optimistic	Central	Pessimistic
HBbVB:	Highly Likely	Certain	Certain
nHBbVB:	Highly Unlikely	Impossible	Impossible

CASE #8:

This case covers sequences in which there is no significant leakage from the primary containment. There is thus little, if any, hydrogen in the reactor building that could burn. The probability of a burn is thus zero.

QUANTIFICATION

	Optimistic	Central	Pessimistic
HBbVB:	Impossible	Impossible	Impossible
nHBbVB:	Certain	Certain	Certain

QUESTION #63: What is the level of reactor building (RB) breach/bypass before VB?

This question determines the impact of possible hydrogen burning on the integrity of the reactor building. If hydrogen burning occurs in the reactor building, rapid pressure increases would occur that could alter gas flow pathways through the building by damaging doors and/or walls or could cause blow-out panels to fail. The burns would also tend to increase the level of bypass by shortening the residence time of the aerosol-carrying gases (less effective retention). Three sources of information have been considered in quantifying this question. The first source is the STCP calculations that indicate that the peak pressure generated by a hydrogen burn in the reactor building far exceeds the design pressure. This result may lead one to believe that a hydrogen burn would destroy the building. Sandia and ORNL calculations, using CONTAIN, HECTRE, and MELCOR with a multivolume volume representation of the reactor building rather than the single volume MARCH (STCP) calculation, indicate relatively benign peak pressures (less than 50% of the design pressure). However, it has been found that these results are highly sensitive to assumptions about the flame speed and the concentration of hydrogen required for ignition. If one selects pessimistic values for these parameters results similar to the STCP calculation can be obtained. The third source of information considered is the opinions of the experts convened to consider containment and source-term issues relevant to the SARRP estimate of risk for Peach Bottom [6]. The consensus of these experts was that the probability of a hydrogen burn exacerbating reactor building bypass was 0.2 (maximum 0.6, minimum 0).

This is a Type 2 question, and there are four branches:

1. nERBB: There is no loss of the reactor building integrity prior to vessel breach.

2. ERBB2: There is significant leakage from the reactor building prior to vessel breach. However, the majority of the gas flow passes through the building with some opportunity for fission product deposition.

3. ERBB3: The reactor building is essentially bypassed prior to vessel breach. The release pathway is either directly to the refueling bay, or damage to the reactor building results in similar levels of fission product retention.

4. ERBB4: The reactor building is completely bypassed. There would be no significant fission product retention.

CASE #1:

In this case, bypass of the reactor building began prior to the opportunity for hydrogen burning. The building thus remains in this condition.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nERBB:	Impossible	Impossible	Impossible
ERBB2:	Impossible	Impossible	Impossible
ERBB3:	Impossible	Impossible	Impossible
ERBB4:	Certain	Certain	Certain

CASE #2:

In this case, either primary containment leakage is from the drywell head (directly to the refueling bay), or the reactor building has been bypassed to the extent that a similar level of fission product retention would be expected. Hydrogen burning has occurred. A hydrogen burn in the refueling bay would almost certainly blow out the blow-out panels. With the panels gone and if the release were directly to the refueling bay, what would be essentially a total bypass would result. For sequences in which the release is not to the refueling bay, a Level 3 reactor building bypass would imply previous structural damage to the building; it is assumed likely that a hydrogen burn would exacerbate this damage.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nERBB:	Impossible	Impossible	Impossible
ERBB2:	Impossible	Impossible	Impossible
ERBB3:	Indeterminate	Unlikely	Impossible
ERBB4:	Indeterminate	Likely	Certain

CASE #3:

In this case, the reactor building is either intact or leaking through the blow-out panels (Level 2 bypass) at the time a hydrogen burn occurs. The central estimate of the probability that the burn causes sufficient damage to the reactor building to result in complete bypass is taken as 0.22; the average value assigned by the SARRP expert review group. If the building does not fail, it is assumed to (continue to) leak. The assumption that a hydrogen burn will always result in at least some uncontrolled leakage is well supported by the computer code calculation described above in the introduction to this question. A factor of 10 uncertainty in the bypass probability is assumed. In the pessimistic case, bypass is assumed certain after a burn.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nERBB:	Impossible	Impossible	Impossible
ERBB2:	Very Likely	Likely	Impossible
ERBB3:	Impossible	Impossible	Impossible
ERBB4:	Very Unlikely	Unlikely	Certain

CASE #4:

In this case, the reactor building previously had Level 3 bypass, and no hydrogen burning has occurred. The bypass level thus remains the same.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nERBB:	Impossible	Impossible	Impossible
ERBB2:	Impossible	Impossible	Impossible
ERBB3:	Certain	Certain	Certain
ERBB4:	Impossible	Impossible	Impossible

CASE #5:

This case includes sequences in which the reactor building had Level 2 bypass, and no hydrogen burns occurred. The bypass level thus remains the same.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nERBB:	Impossible	Impossible	Impossible
ERBB2:	Certain	Certain	Certain
ERBB3:	Impossible	Impossible	Impossible
ERBB4:	Impossible	Impossible	Impossible

CASE #6:

This case covers sequences in which the reactor building was intact and remains so after the possibility of hydrogen burns is considered since none are found to occur. (Either because primary containment is intact, or there was just no burn.)

QUANTIFICATION

	Optimistic	Central	Pessimistic
nERBB:	Certain	Certain	Certain
ERBB2:	Impossible	Impossible	Impossible
ERBB3:	Impossible	Impossible	Impossible
ERBB4:	Impossible	Impossible	Impossible

QUESTION #64: Are fire systems actuated before VB?

This question determines the impact of possible hydrogen burning on the operation of the reactor-building fire sprays. The fire spray system is discussed above (Question #60).

This is a Type 2 question, and there are two branches:

1. EFireS: The reactor-building fire sprays are operating prior to vessel breach.

2. nEFireS: The reactor-building fire sprays are not operating prior to vessel breach.

256

CASE #1:

In this case, the fire sprays were not operating previously, and no hydrogen burn occurred in the secondary containment that would cause them to start. Alternatively, leakage from the primary containment is through the drywell head seal directly to the refueling bay. Hydrogen burns in the refueling bay are assumed not to trigger the fire sprays. Thus, the fire sprays are certain not to operate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
EFireS:	Impossible	Impossible	Impossible
nEFireS:	Certain	Certain	Certain

CASE #2:

This case covers sequences in which either the fire sprays came on regardless of any hydrogen burn in the reactor building or sequences in which there was a hydrogen burn. Hydrogen burns are certain to activate the fire system since they are designed to come on if there is a fire in the reactor building.

QUANTIFICATION

	Optimistic	Central	Pessimistic
EFireS:	Certain	Certain	Certain
nEFireS:	Impossible	Impossible	Impossible

QUESTION #65: Does SGTS operate until VB?

This question determines the impact of possible hydrogen burns in the reactor building on the Standby Gas Treatment System. As noted above (Question #61), fire dampers are expected to isolate the SGTS should the temperature in the reactor building be elevated significantly. This feature is intended to protect the SGTS from fire.

This is a Type 2 question, and there are three branches:

1. EfSGTS: The Standby Gas Treatment System is failed (isolated).

2. ErSGTS: The Standby Gas Treatment System is unavailable but may be recovered.

3. E-SGTS: The Standby Gas Treatment System is operating prior to vessel breach.

BEST COPY AVAILABLE

CASE #1:

In this case, the SGTS was operating prior to the possibility of a hydrogen burn. The fire sprays have not started (i.e., there was no burn). Since fire sprays and SGTS isolation are triggered by the same thing (high reactor-building temperature) the SGTS is thus certain to continue to run.

QUANTIFICATION

	Optimistic	Central	Pessimistic
EfSGTS:	Impossible	Impossible	Impossible
ErSGTS:	Impossible	Impossible	Impossible
E-SGTS:	Certain	Certain	Certain

CASE #2:

In this case, SGTS was not running prior to the possibility of a hydrogen burn, but it was recoverable (i.e., the fire dampers remain open). As with the previous case, the fire sprays are not on so the fire dampers will remain open. SGTS thus remains recoverable.

QUANTIFICATION

	Optimistic	Central	Pessimistic
EfSGTS:	Impossible	Impossible	Impossible
ErSGTS:	Certain	Certain	Certain
E-SGTS:	Impossible	Impossible	Impossible

CASE #3:

This case covers sequences in which the SGTS was previously failed or in which a hydrogen burn occurred. The burn is assumed to both actuate the fire sprays and isolate the SGTS.

QUANTIFICATION

	Optimistic	Central	Pessimistic
EfSGTS:	Certain	Certain	Certain
ErSGTS:	Impossible	Impossible	Impossible
E-SGTS:	Impossible	Impossible	Impossible

QUESTION #66: What is containment pressure after CF or vent during meltdown?

This question determines the containment pressure at the time of vessel breach for those sequences in which the containment has been vented or has failed during core degradation. This question is thus irrelevant for sequences in which either the containment has remained intact or in which it failed prior to core degradation. Containment pressure at the time of vessel breach for sequences in which containment failure occurred prior to core damage is given by Question #36, above. For those sequences with containment failure during vessel breach, this question inputs the containment pressure (in psig) to the event tree code as Parameter #3.

This is a Type 4 question, and there is only one branch:

1. PBase2: This question actually has no branches. The value of Pbase2, the containment failure at vessel breach for sequences in which the containment failed or was vented during core degradation, is merely determined for future reference by the tree.

CASE #1:

In this case, the containment has ruptured or been vented. It is thus depressurized. The pressure will be only slightly greater than atmospheric so Parameter #3 is assigned 20 psia (the precise value is unimportant).

CASE #2:

This case covers sequences in which the containment is intact or has a leak which is of a size which is insufficient to cause depressurization. For sequences in which the containment is intact after core degradation the parameter evaluated by this question (#3) is not referenced. Thus, this case considers sequences in which there was a leakage failure (Level 2 as described in Question #30 and Question #53) during core degradation. The containment pressure is thus slightly below the failure pressure of the containment. The central estimate of mean failure pressure is 132 psia so the evaluation here is 130 psia. The optimistic evaluation is 170 psia. The pessimistic value is 100 psia, the lowest pressure that the SARRP expert group identified [6] as allowing leakage.

BEST COPY AVAILABLE

QUESTION #67: Is there injection to the RPV up to the time of vessel breach?

This question determines whether injection to the RPV continues, either up to the time of RPV meltthrough or until the accident is completely arrested. Because the ECC systems are assumed to fail if the suppression pool is saturated, failure of the containment during core damage may result in these systems becoming inoperable.

This is a Type 2 question, and there are three branches:

1. E5-LPIn: Injection from the low pressure systems (LPCS, LPCI, or HPSW) is maintained through the time of vessel breach (breach may be prevented).
2. E5-HPIn: Injection from the control rod drive hydraulic system is maintained through the time of vessel breach.
3. nE5-In: There is no injection to the RPV at the time of vessel breach.

CASE #1:

In this case, injection to the RPV was restored during core damage. Nothing has occurred subsequently which may hamper injection.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-LPIn:	Certain	Certain	Certain
E5-HPIn:	Impossible	Impossible	Impossible
nE5-In:	Impossible	Impossible	Impossible

CASE #2:

This case also covers scenarios in which there is injection to the RPV during core damage. However, in the case, injection was provided by either the condensate or HPSW pumps. Containment integrity has been lost (venting or failure) during core damage. Because the condensate and HPSW pumps take suction outside containment, a negligible probability of failure as a result of the loss of containment integrity is assumed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-LPIn:	Certain	Certain	Certain
E5-HPIn:	Impossible	Impossible	Impossible
nE5-In:	Impossible	Impossible	Impossible

CASE #3:

In this case, injection to the RPV during core damage was provided by the low-pressure ECC pumps (LPCS or LPCI). These pumps are assumed to fail as a result of containment integrity being lost. However, the CRD pumps are operating and at least that level of injection is maintained. In addition, either the condensate or HPSW pumps are available. Low-pressure injection to the RPV will be maintained if the operators realize that the ECC pumps have failed and align either condensate or HPSW for injection to the RPV. The probability for Branch 1 in this case reflects the assessed probability that the operators will take such action.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-LPIn:	Certain	Very Likely	Likely
E5-HPIn:	Impossible	Very Unlikely	Unlikely
nE5-In:	Impossible	Impossible	Impossible

CASE #4:

This case is similar to the previous one. However, injection from the CRD pumps is not available. Thus, if the operators fail to use the condensate or HPSW systems for injection, there will be no injection to the RPV whatsoever.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-LPIn:	Certain	Very Likely	Likely
E5-HPIn:	Impossible	Impossible	Impossible
nE5-In:	Impossible	Very Unlikely	Unlikely

CASE #5:

In this case, only the CRD pumps were injecting during core degradation. It is assumed that they continue to operate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-LPIn:	Impossible	Impossible	Impossible
E5-HPIn:	Certain	Certain	Certain
nE5-In:	Impossible	Impossible	Impossible

CASE #6:

Two possible scenarios are considered by this case. In the first, injection to the RPV was never restored. In the second, pressurization of the containment during core degradation resulted in reclosure of the SRVs (Question #48). For either scenario, further injection of water to the RPV is precluded.

QUANTIFICATION

	Optimistic	Central	Pessimistic
E5-LPIn:	Impossible	Impossible	Impossible
E5-HPIn:	Impossible	Impossible	Impossible
nE5-In:	Certain	Certain	Certain

QUESTION #68: What is the nature of meltdown?

This question determines whether the core melt proceeds by the gradual flow of molten corium into the lower plenum or by catastrophic failure of the core supports resulting in complete collapse of the core. Despite numerous analyses and various models, there is still no widespread agreement among experts as to how core degradation will progress. This question resolves the disagreement into two distinct possibilities. A slump type of melt was that envisioned in WASH-1400 and propagated into the current MARCH 3 code (some variations on the original slump concept are modeled). The slump model assumes that as the core melts the debris collects on the core plate until the core plate fails (due to heating and/or melting). Failure of the core plate is assumed to result in the entire reactor core "slumping" downward. This model is more applicable to PWRs than BWRs since the BWR core plate provides lateral support only. The core is supported by the control-rod guide tubes in four fuel-bundle groups. However, the source term code package models assume a slump-type failure for BWRs. Other models for BWRs, most notably MELRPI (developed by R. Lahey et al. for the NRC Severe Accident Sequence Analysis program) and the IDCOR program MAAP model, assume that molten fuel can accumulate on the core plate, (or fuel bundle tie plates) and at some point will locally melt the

262

core plate allowing flow of the molten material down the guide tubes into the lower plenum. This flow-type melt is more gradual than the slump type melt. In the flow melt scenario, a vessel bottom head penetration is assumed to melt through before a large quantity of core debris has accumulated in the lower plenum; thus only small quantities (of essentially solid) debris could be ejected were the RPV to fail at high pressure. In the slump scenario, at the time of vessel meltthrough, the lower plenum contains a large quantity of semimolten debris.

This is a Type 2 question, and there are three branches:

1. Slump: The entire core slumps into the lower plenum prior to vessel breach.
2. Flow: Molten material flows from the core region into the lower plenum. Vessel breach may occur with some part of the core intact.
3. nMelt: Core degradation did not occur.

CASE #1:

This case includes sequences in which adequate injection was provided by operator action prior to significant core damage. There is thus no core melt.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Slump:	Impossible	Impossible	Impossible
Flow:	Impossible	Impossible	Impossible
nMelt:	Certain	Certain	Certain

CASE #2:

In this case, injection was restored during core degradation with the high-capacity, low-pressure systems (condensate, LPCS, LPCI, HPSW). Such injection is judged very likely to mitigate gross structural damage to the core-support structure, even if it is unable to prevent the migration of molten core material into the lower plenum of the vessel. Water injection may very well reduce the likelihood of slump type melts. However, since the nature of in-vessel melt progression is not well understood, assignment of varying branch probabilities for the differing cases of this question is not well supported.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Slump:	Impossible	Unlikely	Indeterminate
Flow:	Certain	Likely	Indeterminate
nMelt:	Impossible	Impossible	Impossible

CASE #3:

In this case, there is injection from the CRD hydraulic system only. The STCP calculation of a TC sequence with CRD injection during core degradation [1] predicts that the CRD flow can profoundly influence melt progression. The CRD flow has the effect of cooling the core-support structure (assumed to be the core plate in the STCP) and thus retaining core debris in the core region longer than if there had been no CRD flow (comparing TC2 and TC3 with TC1 in [9]. Since CRD flow can be reasonably expected to decrease the vulnerability (if any) of the core-support structure to collapse, it is judged that it has the potential to increase the probability of a flow-type melt. As in the previous case, a uniform assignment of branch-point probabilities between cases has been assumed since there is virtually no evidence to support the above-stated opinion.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Slump:	Impossible	Unlikely	Indeterminate
Flow:	Certain	Likely	Indeterminate
nMelt:	Impossible	Impossible	Impossible

CASE #4:

This case covers sequences in which there is no delayed restoration of water injection. Quantification of this question is judged to reflect current thought in the reactor-safety community as to whether BWR core melts are more likely to be flow-type melts or slump-type melts. Flow-type melts are considered likely in the central estimate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Slump:	Highly Unlikely	Unlikely	Indeterminate
Flow:	Highly Likely	Likely	Indeterminate
nMELT:	Impossible	Impossible	Impossible

QUESTION #69: What is the mode of vessel breach?

This question determines the nature of the failure of the vessel.

This is a Type 2 question, and there are four branches:

1. Alpha: There is a steam explosion in-vessel with debris forming a missile that subsequently lifts the upper vessel head resulting in failure of the drywell.
2. SE-BH: An in-vessel steam explosion causes catastrophic failure of the vessel bottom head.
3. Mthru: The vessel bottom head melts through as a result of the accumulation of hot core debris on its inner surface.
4. nBreach: Core melt is arrested in-vessel.

CASE #1:

This case covers sequences with no core damage. There can be no vessel breach.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Alpha:	Impossible	Impossible	Impossible
SE-BH:	Impossible	Impossible	Impossible
Mthru:	Impossible	Impossible	Impossible
nBreach:	Certain	Certain	Certain

CASE #2:

In this case, it is assumed that a slump type of core melt occurs (defined in Question #67). This (according to some experts) gives rise to the possibility that a steam explosion could occur. Only the pessimistic walkthrough allows this as a significant possibility. The central estimate reflects the opinion of the Steam Explosion Review Group (SERG) [16].

QUANTIFICATION

	Optimistic	Central	Pessimistic
Alpha:	Impossible	Highly Unlikely	Very Unlikely
SE-BH:	Impossible	Impossible	Unlikely
Mthru:	Certain	Highly Likely	Likely
nBreach:	Impossible	Impossible	Impossible

CASE #3:

This case includes sequences in which injection is restored during core degradation. There is thus some possibility that vessel breach may be avoided. This depends on whether the core debris in the vessel is in a coolable geometry. Just what this means is not clear, and expert opinion varies. Clearly there will be a sensitivity to the extent of core damage at the time injection begins. The IDCOR position is that less than 20% of the core must be melted in order for there to be recovery. Since melt proceeds rapidly once it has begun, there seems little justification for a high probability of preventing vessel failure. It is thus judged that the probability of preventing vessel failure with late injection is indeterminate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Alpha:	Impossible	Impossible	Impossible
SE-BH:	Impossible	Impossible	Impossible
Mthru:	Unlikely	Indeterminate	Likely
nBreach:	Likely	Indeterminate	Unlikely

CASE #4:

This case includes long term station blackout (initial high pressure injection) in which CRD flow only is restored during core degradation. If the operators were able to optimize the flow from this system, flow could approach the level necessary to remove the decay power (9 to 10 hours from scram). However, because core damage is occurring the heat from metal-water reaction would also have to removed. There is thus a slight (very) possibility that vessel breach may be prevented (recall CRD flow is injected through the vessel bottom head).

QUANTIFICATION

	Optimistic	Central	Pessimistic
Alpha:	Impossible	Impossible	Impossible
SE-BH:	Impossible	Impossible	Impossible
Mthru:	Likely	Highly Likely	Certain
nBreach:	Unlikely	Highly Unlikely	Impossible

CASE #5:

This case covers sequences in which a flow-type melt has been assumed. Steam explosions are excluded on the basis that there is insufficient mechanical energy available to trigger an explosion that has enough energy to fail the vessel. In addition, only a small fraction of the core debris is assumed molten at any one time. These sequences are thus assumed to result in a vessel meltthrough.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Alpha:	Impossible	Impossible	Impossible
SE-BH:	Impossible	Impossible	Impossible
Mthru:	Certain	Certain	Certain
nBreach:	Impossible	Impossible	Impossible

QUESTION #70: Is the RPV blown down before significant melt ejection?

This question determines the pressure within the Reactor Pressure Vessel (RPV) at the time core debris is ejected. Note that Question #38 above has previously determined the RPV pressure during core degradation. The purpose of this question is to allow for the possibility that the meltthrough of the vessel (determined by Question #68 and Question #69) occurs in such a way that the gas pressure is relieved before a significant amount of corium is ejected. This could occur if the RPV pressure boundary is broken first in a control rod guide tube at a location that is both above the debris accumulated in the bottom head (if any) and below debris accumulated at the elevation of the core plate. Failure of an instrument tube penetration before a large amount of debris has collected on the bottom head of the vessel could have the same result.

This is a Type 2 question, and there are two branches:

1. nEBlowD: The Reactor Pressure Vessel is at high pressure when core debris begins to be ejected.

2. EBlowD: The Reactor Pressure Vessel is depressurized as a result of localized melting prior to significant debris ejection.

CASE #1:

In this case, core degradation occurs with the RPV at low pressure. The pressure is thus low at vessel breach regardless of the failure mode.

BEST COPY AVAILABLE

QUANTIFICATION

	Optimistic	Central	Pessimistic
nEBlowD:	Impossible	Impossible	Impossible
EBlowD:	Certain	Certain	Certain

CASE #2:

In this case, the core melt is assumed to be of the flow type (described in Question #68 above). The vessel bottom head fails by meltthrough. In a flow melt, corium will flow down the control-rod guide tubes and will accumulate around the stub tubes where the control-rod drive enters the vessel. A leak at this point in the pressure boundary could occur before a significant quantity of core debris had accumulated in the lower plenum. This leak would have the potential for depressurizing the RPV prior to a gross meltthrough of the head. A less likely possibility is that corium flows through the fuel support casting and down the inside of the control rod guide tube. A meltthrough of the pressure boundary inside the guide tube could then occur, isolated from any large quantities of debris which may have accumulated. The latter is less likely since the inside of the guide tube is probably full of water. The central estimate of the probability of vessel depressurization prior to significant melt ejection is 0.6. This value represents the consensus of the group of experts convened to review containment-loading issues for the SARRP evaluation of Peach Bottom risk.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nEBlowD:	Impossible	Indeterminate	Likely
EBlowD:	Certain	Indeterminate	Unlikely

CASE #3:

This case covers sequences in which the vessel is at high pressure during core degradation, and there is a slump-type melt so there is a reduced probability of blowdown prior to a large failure of the bottom head. For slump melts the core debris would have to be essentially solid to permit vessel depressurization prior to significant debris ejection. Since those investigators who support the core-slumping model generally envision the debris to be in at least a partially liquid state, the central estimate is that the probability of early depressurization for this case is negligible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nEBlowD:	Certain	Certain	Certain
EBlowD:	Impossible	Impossible	Impossible

QUESTION #71: Does direct heating occur?

This question determines whether there is direct heating of the containment. If the vessel fails at high pressure, very hot core debris can be blown into the containment atmosphere. If the containment contains significant oxygen or steam, the debris may burn as zirconium, and uranium oxide, oxidize. A portion of the sensible heat of the debris will be transferred to the containment atmosphere with relatively high efficiency. This results in rapid pressurization of the containment. This phenomenon is direct heating. If a significant fraction of the core debris (>10%) participates in direct heating, the containment will almost certainly fail as a result of overpressurization.

This is a Type 2 question, and there are two branches:

1. DH: Direct heating occurs at vessel breach.

2. nDH: There is no direct heating of the primary containment.

CASE #1:

In this case, the vessel pressure is high at the time of vessel breach. The melt progression is assumed to be a core slump so that there are large quantities of molten core debris in the vessel bottom head. These are the conditions that are assumed to allow direct heating to occur. However, the confined geometry of the BWR pedestal, and the restricted opening to the drywell (2 meters by 1 meter) will serve to mitigate the dispersal of debris into the drywell airspace. (It is also noted that the pipes and associated whip restraints, as well as pumps and equipment in the drywell, will also tend to hamper debris dispersal.) Thus, even if the conditions in the RPV at vessel breach favor direct heating, it is not certain to occur.

QUANTIFICATION

	Optimistic	Central	Pessimistic
DH:	Indeterminate	Likely	Certain
nDH:	Indeterminate	Unlikely	Impossible

CASE #2:

This case covers sequences in which either there has been a flow type melt (so little debris has accumulated on the bottom head) or the vessel is depressurized at vessel breach. Either of these conditions is assumed to prevent direct heating.

QUANTIFICATION

	Optimistic	Central	Pessimistic
DH:	Impossible	Impossible	Impossible
nDH:	Certain	Certain	Certain

QUESTION #72: Does an ex-vessel steam explosion occur?

This question determines whether a steam explosion occurs as a result of corium ejection from the RPV. A steam explosion could result from the hot melt falling into water in the sumps or water that has accumulated on the drywell floor. For the purposes of this question, only steam explosions of sufficient magnitude to cause a significant increase in containment pressure are considered. Small steam explosions, which may result in small, incoherent pressure spikes, are more probable but are not included since they have a negligible impact on containment integrity. In a discussion of this issue, M. Corradini (University of Wisconsin) stated that he had recently investigated the depth of water necessary to support a large steam explosion. His results indicate that a few meters are required. At Peach Bottom, a maximum of less than 1 meter is available (in the sumps). The maximum depth on the drywell floor is about half a meter.

This is a Type 2 question, and there are two branches:

1. ExSE: There is a steam explosion in the containment at vessel breach.

2. nExSE: There is no steam explosion in the containment.

CASE #1:

In this case, a slump type melt has occurred. Slump-type core melts are a prerequisite for steam explosions. The slump melt allows for the possibility of a large quantity of hot-molten material to fall from the vessel into the water beneath at the time of vessel breach. If a

270

sufficiently large mass of corium falls into a water pool with enough velocity to induce fragmentation of the molten material, a steam explosion is likely. Three factors mitigate the probability of a steam explosion. First, the amount of material that is molten when the vessel bottom head fails may not be very large or may mix with solid debris prior to reaching the water in the bottom of the pedestal. Second, the control-rod drive mechanisms and their supporting structural beams will inhibit the fall of molten material, and a significant quantity may refreeze on the steel surfaces. Third, apart from water in the sumps, there can be, at most, half a meter of water accumulated on the pedestal floor. This water could be displaced by the falling material, and mixing of the water and corium could be poor. However, it should be noted that the pedestal would tend to contain the water, and while that water might be temporarily displaced, eventual mixing is likely. The above factors were considered in quantifying the probability of an ex-vessel steam explosion in this case. The central estimate is that it is very unlikely (0.01).

QUANTIFICATION

	Optimistic	Central	Pessimistic
ExSE:	Impossible	Very Unlikely	Unlikely
nExSE:	Certain	Very Likely	Likely

CASE #2:

This case covers sequences in which either there is a flow melt or in which core melt is arrested in-vessel. In either case, there is no possibility of a significant steam explosion. If the vessel remains intact, this is clearly true. In the case of a flow melt, corium leaves the vessel more or less as the fuel melts. This is a relatively slow rate. The corium is also close to the solidus temperature and should readily refreeze on the control-rod drive structure below the vessel. With small quantities of solid, or almost solid, debris falling into the pedestal water (if any), there is a negligible probability that a steam explosion, of a magnitude that could impact the containment integrity, would occur.

QUANTIFICATION

	Optimistic	Central	Pessimistic
ExSE:	Impossible	Impossible	Impossible
nExSE:	Certain	Certain	Certain

QUESTION #73: What is pressure rise due to blowdown (effective VS and RCdry)?

This is the first of three questions (Question #73, Question #74, and Question #75) that address the pressure rise in the containment resulting from vessel meltthrough. Each of the questions assigns a value to a separate pressure-rise parameter. The sum of these three parameters is the total pressure rise (in psi) resulting from vessel breach. This question quantifies the pressure rise that results from the blowdown of the vessel. The blowdown of the vessel from 1100 psi, assuming 50% steam by volume, results in approximately a 80 psi rise in the containment pressure. Uncertainties in this value arise from uncertainties in the bulk-gas temperature in-vessel, the hydrogen content, and the final temperature in the containment (considering PV work). The impact of possible direct heating is also considered in this question. Quantification of the pressure rise due to direct heating can be accomplished with a simple model, but the uncertainties are large. These uncertainties are principally the amount of debris that participates in direct heating, the extent of distribution into the containment atmosphere, the rate of debris ejection, and the degree to which the debris undergoes exothermic reactions with the atmosphere. The central estimates for pressure rise for this case were derived on the basis of responses from the group of experts convened to review SARRP containment loading issues for Peach Bottom [6].

This is a Type 4 question, and there is only one branch:

1. DPblow: This question has no actual branches. The purpose of this question is to evaluate Dpblow, the pressure rise in the containment due to blowdown of the RPV. The question is not referenced by subsequent questions, only the evaluated parameter (#4) is used.

CASE #1:

In this case, direct heating occurs. A simple hand calculation of the resultant pressure rise has been made. The central estimate is thus a 96 psi increase, assuming 50% of the core debris participates.

CASE #2:

In this case, the RPV is at high pressure when blowdown occurs. Direct heating does not occur. The central estimate for the pressure rise in containment is taken from the STCP calculation for the TB1 scenario [9].

272

CASE #3:

This case covers sequences in which the vessel is depressurized at the time of vessel breach. There is thus no significant pressure rise in the containment resulting from the equilibration of the pressures inside and outside the vessel.

QUESTION #74: What is the pressure rise due to debris quenching?

This question determines whether there is more water in the base of the reactor pedestal than that contained in the sumps. The pressure rise that results from the core debris contacting that excess water is also evaluated. The value of the pressure rise resulting from the quenching of some fraction of the debris is assigned by this question (Parameter #5). This evaluation is based on the consensus of expert opinion, as in the previous question. The water level in the drywell sumps varies stochastically from essentially zero to the full capacity of approximately 300 cubic feet. The amount at any given time depends on when the operators last decided to pump water out of the sumps. It is assumed that on the average, the sumps are half full. The sumps are assumed to contain 500 gallons for the purpose of evaluating pressure rise due to debris quenching when the drywell floor is "dry" (sump water only). (Subsequant to this evaluation it was realized that the Peach Bottom sumps can contain 2,000 gallons rather than the 1,000 gallons assumed herein.) If containment sprays were operated during the accident, if there is ECC injection at the time of vessel breach, or if the initiator was a primary system break, then the drywell floor is "wet." There may be up to one-half meter of water accumulated.

This is a Type 4 question, and there are two branches:

1. RCwet: The reactor pedestal floor is covered by up to one-half meter of water.
2. RCdry: There is no water in the reactor pedestal beyond what is contained in the drywell sumps.

CASE #1:

In this case, the accident initiator is a primary system break in the drywell. With this leakage from the RPV there will be much more water on the drywell floor than would be present due to condensation in sequences where steam discharge from the RPV is passed to the suppression pool through the SRVs. The drywell floor is thus "wet," but there is probably less water accumulated than if containment spray had operated or if ECC injection occurs. The optimistic and central estimates of the pressure rise (7 and 15 psi respectively) are thus 50% of the values used for sequences in which sprays operated. The pessimistic value is identical, since the pessimistic assumption would be that there is no difference.

QUANTIFICATION

	Optimistic	Central	Pessimistic
RCwet:	Certain	Certain	Certain
RCdry:	Impossible	Impossible	Impossible

CASE #2:

In this case,either the drywell sprays operated at some time during the accident, or there was injection of water to the vessel prior to vessel breach; the injection is assumed to continue. The drywell floor is thus "wet." The central estimate of the pressure rise due to debris quenching is 30 psi. The optimistic and pessimistic estimates reflect the range of opinions given by the experts and assigns a plus or minus 50% uncertainty.

QUANTIFICATION

	Optimistic	Central	Pessimistic
RCwet:	Certain	Certain	Certain
RCdry:	Impossible	Impossible	Impossible

CASE #3:

In this case, there is no water on the drywell floor. The only water available to quench debris is the 500 gallons assumed to be in the drywell sumps. The central estimate of the pressure rise due to debris quenching in the sumps is 10 psi. The optimistic estimate is 5 psi; the lowest value given any credence by the experts. The pessimistic value is 30 psi; the largest value given any credence by the SARRP expert reviewers [6].

QUANTIFICATION

	Optimistic	Central	Pessimistic
RCwet:	Impossible	Impossible	Impossible
RCdry:	Certain	Certain	Certain

QUESTION #75: Is SP bypassed or vents not cleared at vessel breach of vapor suppression--DPfVS

This question is the last of the three that are used to assess the containment pressure rise at vessel breach. The pressure rise indicated in the two previous questions was evaluated assuming that pressure suppression occurs. This question adds a pseudo-pressure rise that accounts for suppression pool bypass, pool temperatures close to the saturation point, and dynamic loads resulting from steam explosions and direct heating. Thus, the parameter evaluated below is a correction to the previously evaluated pressure increases to account for any lack of effectiveness in the pressure-suppression system. Because the question has branches and evaluates the pressure-rise parameter, there can be "cases within a case." The value that the pressure-rise parameter (#6) assumes is dependent on which question branch is taken in that case. Thus the probability that the vents have time to clear in a specific case will impact the predicted pressure-rise. Evaluation of the pressure-rise parameter for cases in which there are ex-vessel steam explosions is based on hand calculations performed at SNL that attempted to extrapolate results from MARCH calculations made prior to the development of the STCP. These estimates assume that the pedestal contains 2 feet of water that is instantaneously vaporized at vessel breach. The drywell vents are assumed not to clear (i.e., the drywell alone must contain the steam produced). These assumptions probably overestimate the pressure produced, even though hygrogen generation has been neglected. However, in the current quantification of the tree, the numerical values, provided that they are greater than about 50 psi (in the central estimate), have no impact on the results of the containment event tree analysis.

This is a Type 4 question, and there are three branches:

1. Vclr: The drywell vents clear (no large dynamic load), and the pressure-suppression system works as designed.

2. nVclr: There is a large, rapid increase in the drywell pressure, and the drywell vents do not clear with sufficient rapidity.

3. Bypass: The suppression pool is bypassed, and there is no vapor-suppression function.

CASE #1:

In this case, there is an ex-vessel steam explosion but no direct heating. The suppression pool is not bypassed, but its temperature is high enough that the effectiveness of vapor suppression is reduced. Because a steam explosion would produce a large dynamic load, there is some probability that the drywell vents will not clear. The resulting pressure rise in the drywell would be large since the steam generated by the explosion would not be condensed. The value assigned to the pressure-rise parameter, given that the vents do not clear, is large enough to ensure containment failure in the central and pessimistic estimates. The optimistic estimate is 70 psi. If the vents do clear, the central estimate of the pressure rise is 50 psi. The optimistic and pessimistic evaluations reflect a plus or minus 50% uncertainty in the central estimate. Since no information is available, the probability that the vents will clear is indeterminate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Likely	Indeterminate	Indeterminate
nVclr:	Unlikely	Indeterminate	Indeterminate
Bypass:	Impossible	Impossible	Impossible

CASE #2:

This case is similar to the previous one, with the exception that the suppression pool has a substantial amount of subcooling. Pressure suppression will thus be effective if the vents clear. The probability that the vents clear is evaluated exactly as in the previous case. When the vents do not clear, the pressure rise is evaluated identically to the previous case. If the vents do clear, the pressure rise is taken to be 5 psi lower than the previous case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Likely	Indeterminate	Indeterminate
nVclr:	Unlikely	Indeterminate	Indeterminate
Bypass:	Impossible	Impossible	Impossible

CASE #3:

In this case, direct heating occurs, there is no ex-vessel steam explosion, and the suppression pool is not bypassed but the water in it is hot. Direct heating can result in a very rapid increase in the containment pressure. If the vents do not clear, the pressure rise is set to 999 psi (this dummy value ensures containment failure). If the

276

vents do clear, the pressure increment required is that due to the suppression pool being saturated rather than being subcooled. Since the containment pressure rise during direct heating is due to the thermal expansion of gas in the drywell and the production of noncondensable gasses, the pressure-suppression function of the pool (i.e., steam condensation) is relatively unimportant. If the pool is saturated, more steam would be produced as a result of hot gases being forced through the pool, giving up their heat. The central estimate is that the additional steam would create a pressure increase of 10 psi in the containment. The optimistic and pessimistic evaluations represent an uncertainty of plus or minus 50%. The probability of the vents failing to clear is based on the responses provided by the expert reviewers. The central estimate of the probability of the vents not clearing is 0.16. This is the average weight (weights were between zero and unity, and are interpreted here to be subjective probabilities) assigned by the experts to the scenario that direct heating results in certain containment failure. The optimistic estimate reduces this probability by a factor of 10. For the pessimistic evaluation, the probability assigned is 0.5, since no analysis is available.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Very Likely	Likely	Indeterminate
nVclr:	Very Unlikely	Unlikely	Indeterminate
Bypass:	Impossible	Impossible	Impossible

CASE #4:

This case is similar to the previous one except that the suppression pool is significantly subcooled. Provided that the vents clear, there is no additional containment pressure rise. The total pressure rise is given in Question #73. If the vents do not clear, containment failure is certain (parameter set to dummy value of 999 psi). The probability that the vents do not clear is quantified as in the previous case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Highly Likely	Likely	Indeterminate
nVclr:	Highly Unlikely	Unlikely	Indeterminate
Bypass:	Impossible	Impossible	Impossible

CASE #5:

In this case, both direct heating and a steam explosion occur at vessel breach. Such an occurrence is extremely unlikely; as indicated by previous quantification of the tree. The suppression pool is also hot so the effectiveness of pressure suppression is reduced. With both a steam explosion and direct heating the drywell will pressurize very rapidly. As in the previous case, if the vents do not clear, the pressure rise is assigned a value so large that containment failure is assured (999 psi). Given that the vents clear the pressure rise is simply the sum of those given in Cases #1 and #3 above. The central estimate for the probability of the vents not being able to clear is taken to be identical to that for Case #1 on the basis that the explosion generates the most rapid increase in pressure and will thus govern the ability of the vents to clear.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Likely	Indeterminate	Indeterminate
nVclr:	Unlikely	Indeterminate	Indeterminate
Bypass:	Impossible	Impossible	Impossible

CASE #6:

This case is similar to the previous one, with the exception that the suppression pool is subcooled and the steam released to, or generated in, the drywell is largely condensed. There is thus a negligible pressure rise from this steam, provided that the drywell vents clear. The pressure-rise parameter is evaluated exactly as in Case #5 on the basis of the same arguments relative to dynamic pressure loading. The probability that the vents clear is evaluated exactly as in the previous case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Likely	Indeterminate	Indeterminate
nVclr:	Unlikely	Indeterminate	Indeterminate
Bypass:	Impossible	Impossible	Impossible

CASE #7:

In this case, there is a significant amount of water on the drywell floor. The RPV was at high pressure at the time of vessel breach but no steam explosion occurred, and direct heating did not occur. The suppression pool temperature is high, but it has not been bypassed. Interaction of the debris with the water on the drywell floor will produce steam and hydrogen in addition to what is released from the vessel. The effectiveness of pressure suppression is reduced by the hot pool. It is certain that the vents will clear (the system is designed for very rapid pressurization of the drywell associated with vapor generation). The pressure-rise parameter for this case accounts for the effect of additional steam generated from the suppression pool in absorbing the steam and hydrogen released by the blowdown of the vessel and generated by the interaction of debris with the water on the drywell floor. The central estimate for the additional pressure rise for this case is 16 psi. This value was determined, assuming a pool temperature of approximately 250°F (180°F for non-saturated) and that 10% of the steam generated would not be condensed in the pool. The pessimistic estimate assumes 35% of the generated steam will not be condensed. Optimistically, all the steam is condensed, and thus the pressure increment is due only to sensible heat (3 psi).

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Certain	Certain	Certain
nVclr:	Impossible	Impossible	Impossible
Bypass:	Impossible	Impossible	Impossible

CASE #8:

This case considers the remaining scenarios in which the suppression pool is not bypassed. All cases in which significant pressure increase (beyond that considered in Question #73 and Question #74) could occur with pressure suppression available, have been considered above. Therefore, this case assigns zero (i.e., negligible) additional pressure rise.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Certain	Certain	Certain
nVclr:	Impossible	Impossible	Impossible
Bypass:	Impossible	Impossible	Impossible

CASE #9:

In this case, and those which follow, the suppression pool has been bypassed. However, in this case the containment has already ruptured. Thus, a zero pressure increase is assigned.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Impossible	Impossible	Impossible
nVclr:	Impossible	Impossible	Impossible
Bypass:	Certain	Certain	Certain

CASE #10:

This case considers direct heating in a dry drywell with no pressure suppression. Based on a hand calculation which assumes 50% of the core mass participates in direct heating, the central estimate pressure rise is approximately 170 psi. A 70 psi correction (to the value in Question #73) to the pressure rise for direct heating is thus assigned.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Impossible	Impossible	Impossible
nVclr:	Impossible	Impossible	Impossible
Bypass:	Certain	Certain	Certain

CASE #11:

This case is similar to the previous one except that there is water on the drywell floor. An even larger pressure rise is assigned based on an estimate of the additional pressure produced by steam production. A value of 105 psi is used for the central estimate. The precise value is unimportant since containment failure is assured.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Impossible	Impossible	Impossible
nVclr:	Impossible	Impossible	Impossible
Bypass:	Certain	Certain	Certain

280

CASE #12:

In this case, the RPV fails at high pressure and the drywell floor is wet. As for all cases after number 8, there is no pressure suppression. A large pressure increase is appropriate, the estimate is greater than 220 psi, assuming all the water in the drywell evaporates. An additional pressure rise of 120 psi is used for this case to approximate that total pressure increase. Again, the precise value is unimportant.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Impossible	Impossible	Impossible
nVclr:	Impossible	Impossible	Impossible
Bypass:	Certain	Certain	Certain

CASE #13:

This case is similar to the previous one with the exception that there is no blowdown of the RPV. Again, all the water on the drywell floor was assumed to be vaporized. A 90 psi value was used to account for the additional pressurization resulting from supression pool bypass.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Impossible	Impossible	Impossible
nVclr:	Impossible	Impossible	Impossible
Bypass:	Certain	Certain	Certain

CASE #14:

In this case, the RPV is depressurized at meltthrough and the drywell floor is dry. Because the suppression pool is bypassed, there is a small additional pressure rise in containment resulting from the steam produced by boil-off of water in the sumps not being condensed. A value of 15 psi is assessed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Impossible	Impossible	Impossible
nVclr:	Impossible	Impossible	Impossible
Bypass:	Certain	Certain	Certain

281

CASE #15:

This case considers the addition pressure increase, that results without pressure suppression due to RPV blowdown combined with evaporation of water in the drywell sumps. A value of 45 psi is used for the central estimate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Vclr:	Impossible	Impossible	Impossible
nVclr:	Impossible	Impossible	Impossible
Bypass:	Certain	Certain	Certain

QUESTION #76: Does pressurization fail containment at VB?

This question instructs the event tree code to determine the probability of containment failure immediately after (at) vessel breach. To do this, the code calculates the containment pressure using parameter values supplied by previous questions (Question #36 and Question #42, or Question #66, and Questions #73 through Question #75). The calculated pressure is used to evaluate the probability of failure based on an input containment fragility. This question is analogous to Question #50 above; the same fragilities are assumed. It must be acknowledged that the simple summation of pressure increments is not an entirely satisfactory means of estimating the pressure in containment that accompanies failure of the RPV. There may be interactions among the mechanisms that produce pressure increases that could either raise or lower the maximum pressure attained. Treatment of the pressure increment according to case, as has been done in the preceding questions, is believed to have eliminated significant sources of this kind of error. The possibility that combinations of phenomena may alter the probability of containment failure from what is calculated by this question does exist.

This is a Type 6 question, and there are two branches:

1. IP: The containment fails at vessel breach.

2. nIP: The containment retains its integrity following vessel breach.

CASE #1:

In this case, the containment has not been vented and did not fail during core degradation. The pressure in the containment immediately prior to vessel breach is "Pbasel," which is evaluated in Question #36, plus the pressure rise due to hydrogen production in-vessel (Question #42). The containment pressure after vessel breach is evaluated by adding the above-defined pressure-rise parameters (Questions #73 through Question #75) to this initial value.

CASE #2:

In this case, the vessel does not meltthrough so the probability of containment failure is assessed based on the containment pressure evaluated by summing the pressure at the onset of core degradation (Question #36) and the pressure rise due to hydrogen production (Question #42). The result is the containment pressure when the accident is recovered (assuming that it did not fail).

CASE #3:

In this case, the containment was previously failed or has been vented. The pressure after vessel breach is thus evaluated using the base pressure given by Question #66, which accounts for containment failure or venting during core degradation. The pressure rise in containment following vessel breach is added (Questions #73 through Question #75). This allows for what may have previously been Level 2 leakage (defined under Question #30) to be exacerbated to a rupture.

QUESTION #77: What is the base pressure before vessel breach?

This question and the one that follows it differ significantly from the other questions in the tree. All the other questions provide a means for the analyst to supply information about various aspects of the accident progression to the analysis; this is done in the assignment of branch-point probabilities and parameter values. These questions function in almost the reverse manner: they supply information about the results of the accident progression back to the analysts. This question provides an estimate of the pressure in the containment immediately prior to vessel breach. This estimate is used by the RELTRAC source-term code to estimate the rate of containment pressurization and the time of containment failure. It was necessary to put this question into the tree because when EVNTRE is run in sampling mode, this is the best way to

283

assure that the sampled values of pressure increments are correctly relayed into the source-term estimate. Each branch of the question corresponds to a range of pressures. The particular branch "taken" for this question can be referenced by RELTRAC. Thus this question is an important part of the interface between the containment and source-term analyses. EVNTRE assigns the branch "taken" in this question by summing the parameters that determine containment pressure prior to vessel breach (1 or 2, and 3) and comparing the result with the value associated with each branch.

This is a Type 6 question, and there are eight branches:

1. PB130: The containment pressure at vessel breach is at least 130 psia.
2. PB110: The containment pressure at vessel breach is greater than or equal to 110 psia but less than 130 psia.
3. PB90: The containment pressure at vessel breach is greater than or equal to 90 psia but less than 110 psia.
4. PB70: The containment pressure at vessel breach is greater than or equal to 70 psia but less than 90 psia.
5. PB50: The containment pressure at vessel breach is greater than or equal to 50 psia but less than 70 psia.
6. PB40: The containment pressure at vessel breach is greater than or equal to 40 psia but less than 50 psia.
7. PB30: The containment pressure at vessel breach is less than or equal to 30 psia but less than 40 psia.
8. PB20: The containment pressure at vessel breach is less than 30 psia.

CASE #1:

In this case, the containment is intact prior to RPV breach. The pressure in containment prior to vessel breach is evaluated based on the sum of the pressure prior to core damage (Question #36) and the pressure increase during core damage (Question #42).

284

CASE #2:

In this case, either the containment has ruptured or the accident has been arrested prior to RPV meltthrough. The logic for this case forces the pressure rise indication to be less that 30 psi (Branch 8). This is clearly true if the containment has ruptured. For those accident progression in which there is no RPV meltthrough this branch is assigned so that the drywell-to-wetwell flows are correctly calculated in RELTRAC.

CASE #3:

In this case, the containment leaks (Level 2) prior to vessel breach. The pressure prior to vessel breach is thus equal to the mean failure pressure, as established by Question #66, Case #2.

QUESTION #78: What is the pressure rise at vessel breach?

The purpose of this question is similar to that of Question #77. This question determines (for RELTRAC) the rise in containment pressure that accompanies vessel breach. EVNTRE accomplishes this by summing the relevant parameters (4, 5, and 6) and comparing the result with the value associated with each branch of this question.

This is a Type 6 question, and there are eight branches:

1. DP130: The containment pressure rise at vessel breach is at least 130 psi.

2. DP110: The containment pressure rise at vessel breach is greater than or equal to 110 psi but less than 130 psi.

3. DP90: The containment pressure rise at vessel breach is greater than or equal to 90 psi but less than 110 psi.

4. DP70: The containment pressure rise at vessel breach is greater than or equal to 70 psi but less than 90 psi.

5. DP50: The containment pressure rise at vessel breach is greater than or equal to 50 psi but less than 70 psi.

6. DP40: The containment pressure rise at vessel breach is greater than or equal to 40 psi but less than 50 psi.

7. DP30: The containment pressure rise at vessel breach is greater than or equal to 30 psi but less than 40 psi.

8. DP20: The containment pressure rise at vessel breach is less than 30 psi.

CASE #1:

In this case, the containment is intact prior to RPV breach. The pressure rise which results from vessel breach is evaluated based on the sum of the pressure due to blowdown of the RPV (Question #73), the pressure due to the interaction between core debris and water in the drywell (Question #74), and a correction to the total of the preceding two that accounts for suppression pool bypass, drywell vents not clearing, and the suppression pool temperature being high (Question #75).

CASE #2:

In this case, either the containment has ruptured in the drywell prior to vessel breach or the accident has been arrested in-vessel. The minimum pressure rise is indicated for this case. This has been done so that RELTRAC correctly calculates flows in the containment.

CASE #3:

In this case, the containment has developed Level 2 leakage prior to vessel breach. This case is quantified identically with Case #1.

QUESTION #79: Does direct melt-structure attack fail containment at VB?

This question determines whether containment failure occurs as a result of core debris contacting the steel containment shell at the point of embedment. Core debris may flow from the vessel pedestal and spread over the drywell floor to the point at which the steel containment shell is embedded in the concrete of the drywell floor and basemat. Two failure modes can be conceived. The debris may be sufficiently hot that the carbon steel of the shell is melted. Or the combined effects of high containment pressure and local heating of the steel shell may result in a creep-rupture type failure. IDCOR (work performed by Chicago Bridge and Iron) has evaluated the latter mode and concluded that the shell is adequately restrained to preclude this mode of failure [7]. It should be noted, however, that this evaluation was carried out assuming a 1500°F debris temperature, whereas a temperature close to, or exceeding, the melting point of the steel would seem more appropriate. It is conceivable that a cooler crust might still form on the debris preventing a meltthrough. However, meltthrough of the containment is clearly possible (especially assuming the CORCON-VANESA model for core-concrete interaction, which can, under some conditions, predict very high debris temperature >2000 degrees C). The factor most likely to limit containment failure by melt-structure attack is the failure of the corium to flow out as far as the containment shell. This failure mode is considered equivalent to a rupture of the drywell at any location below the head seal.

This is a Type 2 question, and there are two branches:

1. IM: The containment is failed as a result of meltthrough at the point of embedment.

2. nIM: The containment is not failed as a result of melt-structure attack.

CASE #1:

In this case, the RPV is not breached. Since core debris never reaches the drywell floor, contact between the debris and the drywell shell is precluded.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IM:	Impossible	Impossible	Impossible
nIM:	Certain	Certain	Certain

CASE #2:

In this case, the core debris was dispersed at vessel breach as a result of the high pressure in the RPV. Such dispersal is judged to slightly increase the probability of containment failure due to meltthrough. The assumption here is that the debris is ejected out of the pedestal manway with considerable velocity directed at the drywell wall. The evaluation of the drywell meltthrough probability is based on the consensus opinion of the containment experts who were convened to consider the SARRP Peach Bottom issues [6]. That probability is 0.95 (central).

QUANTIFICATION

	Optimistic	Central	Pessimistic
IM:	Unlikely	Very Likely	Certain
nIM:	Likely	Very Unlikely	Impossible

CASE #3:

In this case, the in-vessel melt progression was a slump-type melt (defined by Question #68). This model for the in-vessel melt progression presumes that a large fraction of the core (50% or more) flows from the vessel following meltthrough (high-pressure ejection of core debris was considered in the preceding case). There was general agreement among the SARRP experts that, given that this large quantity of debris flowed onto the drywell floor at vessel breach, meltthrough of the drywell shell was likely. A few of the experts were adamant in their belief that it was a certainty.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IM:	Unlikely	Likely	Certain
nIM:	Likely	Unlikely	Impossible

CASE #4:

In this case, the core debris has been dynamically dispersed as a result of an in-vessel steam explosion that failed the bottom head. This dispersal is judged similar to Case #2, in which there is high-pressure debris ejection. A slump-type melt in-vessel is also required for this kind of interaction to occur since there must be a sufficient supply of molten corium at the time of vessel breach. The probability of drywell meltthrough is quantified exactly as in Case #2 of this question.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IM:	Unlikely	Very Likely	Certain
nIM:	Likely	Very Unlikely	Impossible

CASE #5:

This case covers sequences in which there is a flow type in-vessel melt progression (Question #68) and the drywell floor has excess water. Thus, for this case one may envision partially solidified debris falling from the CRD

QUANTIFICATION

	Optimistic	Central	Pessimistic
IM:	Unlikely	Very Likely	Certain
nIM:	Likely	Very Unlikely	Impossible

CASE #3:

In this case, the in-vessel melt progression was a slump-type melt (defined by Question #68). This model for the in-vessel melt progression presumes that a large fraction of the core (50% or more) flows from the vessel following meltthrough (high-pressure ejection of core debris was considered in the preceding case). There was general agreement among the SARRP experts that, given that this large quantity of debris flowed onto the drywell floor at vessel breach, meltthrough of the drywell shell was likely. A few of the experts were adamant in their belief that it was a certainty.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IM:	Unlikely	Likely	Certain
nIM:	Likely	Unlikely	Impossible

CASE #4:

In this case, the core debris has been dynamically dispersed as a result of an in-vessel steam explosion that failed the bottom head. This dispersal is judged similar to Case #2, in which there is high-pressure debris ejection. A slump-type melt in-vessel is also required for this kind of interaction to occur since there must be a sufficient supply of molten corium at the time of vessel breach. The probability of drywell meltthrough is quantified exactly as in Case #2 of this question.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IM:	Unlikely	Very Likely	Certain
nIM:	Likely	Very Unlikely	Impossible

CASE #5:

This case covers sequences in which there is a flow-type in-vessel melt progression (Question #68) and the drywell floor has excess water. Thus, for this case one may envision partially solidified debris falling from the CRD zero simply as a convenience for subsequent logic in the tree.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IM:	Impossible	Impossible	Impossible
nIM:	Certain	Certain	Certain

QUESTION #80: What is the location of containment failure after VB?

This question determines the location in the containment from which leakage (if any) occurs. This question is subsequently used to evaluate the extent of suppression pool and/or reactor building bypass. Question #30 and Question #51 above are analogous.

This is a Type 2 question, and there are four branches:

1. nICL : There is no leakage from the containment up to the time at which core debris is released from the vessel (or core melt is arrested).
2. ICLD: Containment leakage is from the drywell.
3. ICLW: Containment leakage is from the wetwell (torus).
4. ICLDW: Overpressure failure of the containment occurred at vessel breach in the wetwell. The drywell shell was subsequently melted.

CASE #1:

This case considers four possible failure modes each of which is certain to result in primary containment leakage from the drywell: 1) Alpha mode in-vessel steam explosion that lifts the drywell head, 2) meltthrough of the drywell wall not preceded by overpressure failure (previous question), 3) overpressure failure at vessel breach with the drywell vents not clearing (containment fails before the wetwell experiences a significant pressure increase), and 4) previous leakage from the drywell.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Impossible	Impossible	Impossible
ICLD:	Certain	Certain	Certain
ICLW:	Impossible	Impossible	Impossible
ICLDW:	Impossible	Impossible	Impossible

290

BEST COPY AVAILABLE

CASE #2:

In this case, overpressure failure of the containment occurs at RPV meltthrough and drywell meltthrough occurs subsequently. Thus, leakage from the containment following vessel breach is ultimately from the drywell. This case is included so that this detail of the containment failure mode can be passed to the source term calculation. If the containment fails in the torus at vessel breach, the drywell and RPV contents will be purged through the suppression pool until meltthrough occurs. Significant retention of fission products in the pool may result. Previous wetwell failure is signified by Branch 4 of this question. The probability for Branch 4 is identical to the probability of Branch 3 in Question #28, Case 4.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Impossible	Impossible	Impossible
ICLD:	Indeterminate	Likely	Certain
ICLW:	Impossible	Impossible	Impossible
ICLDW:	Indeterminate	Unlikely	Impossible

CASE #3:

In this case, the containment previously leaked from the wetwell. There is no meltthrough of the drywell shell. It is judged certain that leakage will continue from this location.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Impossible	Impossible	Impossible
ICLD:	Impossible	Impossible	Impossible
ICLW:	Certain	Certain	Certain
ICLDW:	Impossible	Impossible	Impossible

CASE #4:

In this case, the containment failed due to overpressurization at vessel breach. There is no meltthrough of the drywell shell. In the absence of large dynamic loads to the drywell, there is some possibility that failure may occur in the wetwell. This case is quantified exactly as Case #4 of Question #28.

BEST COPY AVAILABLE

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Impossible	Impossible	Impossible
ICLD:	Indeterminate	Likely	Certain
ICLW:	Indeterminate	Unlikely	Impossible
ICLDW:	Impossible	Impossible	Impossible

CASE #5:

In this case, the containment remains intact following vessel breach. It is therefore certain that there will be "no significant leakage."

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Certain	Certain	Certain
ICLD:	Impossible	Impossible	Impossible
ICLW:	Impossible	Impossible	Impossible
ICLDW:	Impossible	Impossible	Impossible

QUESTION #81: For drywell failures, is the failure at the head seal?

This question is completely analogous to Question #29 and Question #52. The discussion under Question #29 is relevant.

This is a Type 2 question, and there are three branches:

1. IDHF: Drywell leakage is from the head seal to the refueling bay following vessel breach.

2. IDKF: Drywell leakage is from the knuckle region (or equivalent) following vessel breach.

3. nIDF: The drywell retains its integrity following vessel breach.

CASE #1:

In this case, the drywell wall is melted through at the point of embedment. This failure mode is assumed to be equivalent to overpressure failure at the drywell knuckle.

292

QUANTIFICATION

	Optimistic	Central	Pessimistic
IDHF:	Impossible	Impossible	Impossible
IDKF:	Certain	Certain	Certain
nIDF:	Impossible	Impossible	Impossible

CASE #2:

In this case, leakage was previously at the vessel head. Provided the drywell wall did not meltthrough (previous case), leakage is assumed certain to continue to be predominantly through the head seal.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IDHF:	Certain	Certain	Certain
IDKF:	Impossible	Impossible	Impossible
nIDF:	Impossible	Impossible	Impossible

CASE #3:

There was previous leakage from the knuckle location (or equivalent). Leakage is assumed certain to continue from this location.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IDHF:	Impossible	Impossible	Impossible
IDKF:	Certain	Certain	Certain
nIDF:	Impossible	Impossible	Impossible

CASE #4:

In this case, the containment failed in the drywell as a result of overpressurization at vessel breach. The split between leakage from the head seal and knuckle location is quantified exactly as in Case #3 of Question #29.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IDHF:	Unlikely	Indeterminate	Likely
IDKF:	Likely	Indeterminate	Unlikely
nIDF:	Impossible	Impossible	Impossible

CASE #5:

This case covers sequences in which leakage is predominantly from the wetwell or in which the containment retains its integrity following vessel breach.

QUANTIFICATION

	Optimistic	Central	Pessimistic
IDHF:	Impossible	Impossible	Impossible
IDKF:	Impossible	Impossible	Impossible
nIDF:	Certain	Certain	Certain

QUESTION #82: What is the containment leakage level after VB?

This question determines the level of containment leakage immediately after vessel breach. Given that the containment is no longer intact, two levels of leakage are considered. Level 2 leakage is a failure that results in only slow depressurization of the containment (on the order of 2 hours to reach atmospheric pressure). Level 2 leakage corresponds to a leakage area on the order of 0.1 square foot. Level 3 leakage is a rupture of the containment (several square feet of area) that results in containment depressurization within several minutes.

This is a Type 2 question, and there are three branches:

1. nICL: The containment retains its integrity immediately following vessel breach.

2. ICL2: The containment to leaks subsequent to vessel breach. The leakage rate results in very slow containment depressurization.

3. ICL3: The containment ruptures, resulting in rapid depressurization, subsequent to vessel breach.

CASE #1:

In this case, there is either an Alpha failure (WASH-1400 type in-vessel steam explosion), or the core debris melts the containment at the point of embedment, or the drywell vents did not clear at the time of vessel breach. The Alpha-mode failure results in a missile; the drywell head is assumed to be lifted. Alpha mode is certainly Level 3 leakage. It is assumed that if the drywell is melted at the point of embedment, the leakage area is large enough to result in rapid, depressurization. Drywell failure at the time of vessel breach coupled with the drywell vents being unable to clear is also assumed to result in containment rupture since the pressurization is rapid, and typically peak pressures are well in excess of the assessed ultimate capability of the containment. Sequences in which the containment previously ruptured are also included in this case. The rupture is certain to remain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Impossible	Impossible	Impossible
ICL2:	Impossible	Impossible	Impossible
ICL3:	Certain	Certain	Certain

CASE #2:

This case covers sequences in which the containment failed at vessel breach either at the drywell knuckle (or equivalent location in the drywell) or in the wetwell (torus). This case is quantified exactly as Case #6 of Question #30.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Impossible	Impossible	Impossible
ICL2:	Likely	Unlikely	Impossible
ICL3:	Unlikely	Likely	Certain

CASE #3:

In this case, containment failure at vessel breach resulted in leakage from the drywell head seal. Provided the failure was not induced by sudden pressurization (considered in Case #1 of this question), this failure mode is always assumed to result in Level 2 leakage. This is consistent with Case #7 of Question #30.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Impossible	Impossible	Impossible
ICL2:	Certain	Certain	Certain
ICL3:	Impossible	Impossible	Impossible

CASE #4:

In this case, Level 2 leakage developed prior to vessel breach, and this was not exacerbated at vessel breach. Thus the leakage is certain to remain Level 2.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Impossible	Impossible	Impossible
ICL2:	Certain	Certain	Certain
ICL3:	Impossible	Impossible	Impossible

CASE #5:

This case covers sequences in which there has been no significant threat to the containment integrity up to the time the core debris is released from the vessel. (Sequences in which there is no vessel breach are also included.) There is thus no containment failure, and "no leakage" is certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nICL:	Certain	Certain	Certain
ICL2:	Impossible	Impossible	Impossible
ICL3:	Impossible	Impossible	Impossible

QUESTION #83: Is the suppression pool drained following vessel breach?

This question considers the possibility that containment failure in the torus results in the suppression pool being drained. This question is entirely analogous to Question #31 and Question #57.

This is a Type 2 question, and there are three branches:

1. I-SPD: The suppression pool has been essentially drained as a result of failure below the water level combined with a failure of the torus room to contain the water escaping the pool.
2. IpSPD: The suppression pool level has decreased, as a result of a failure in the torus, such that the downcomers from the drywell vent ring headers are essentially uncovered.
3. InSPD: The suppression pool maintains at least normal operating level.

CASE #1:

In this case, the suppression pool has previously been drained. It is certain to remain in that condition.

QUANTIFICATION

	Optimistic	Central	Pessimistic
I-SPD:	Certain	Certain	Certain
IpSPD:	Impossible	Impossible	Impossible
InSPD:	Impossible	Impossible	Impossible

CASE #2:

In this case, the drywell vents were previously uncovered. It is certain that they remain uncovered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
I-SPD:	Impossible	Impossible	Impossible
IpSPD:	Certain	Certain	Certain
InSPD:	Impossible	Impossible	Impossible

CASE #3:

In this case the torus failed as a result of containment overpressurization at vessel breach. The probability of the suppression pool being drained is quantified exactly as in Case #1 of Question #31.

QUANTIFICATION

	Optimistic	Central	Pessimistic
I-SPD:	Impossible	Very Unlikely	Unlikely
IpSPD:	Very Unlikely	Indeterminate	Likely
InSPD:	Very Likely	Indeterminate	Unlikely

CASE #4:

This case covers scenarios in which there has been no structural failure of the torus below the pool level. The suppression pool thus remains at, or above, normal level.

QUANTIFICATION

	Optimistic	Central	Pessimistic
I-SPD:	Impossible	Impossible	Impossible
IpSPD:	Impossible	Impossible	Impossible
InSPD:	Certain	Certain	Certain

QUESTION #84: Is ac power not available late?

This question allows consideration of offsite power being restored subsequent to vessel breach. Quantification of the probabilities is done using the same data as were used in the ASEP quantification of the blackout sequence frequencies. National data for the time for recovery of offsite power were used. Values for the optimistic and pessimistic estimates for recovery probability were obtained as described under Question #32.

This is a Type 2 question, and there are two branches:

1. LfAC: Ac power is not recovered during the sequence until after most of the fission-product release has occurred.

2. L-AC: Ac power is recovered subsequent to vessel breach and in time to impact the core-concrete interaction.

CASE #1:

In this case, there was initial operation of the high pressure make-up systems, and thus, vessel breach occurs approximately 11 hours after the loss of offsite power. Station batteries are drained (indicated in the logic for this case by the fact that the RPV repressurized prior to core damage). The probability for ac power availability thus approximates the probability of offsite power being available at 12.5 hours (somewhat arbitrary) following interruption, given that it was not available at 11 hours following interruption. As for Case #1 of question #32, the power nonrecovery probability was adjusted to account for loss of dc power used to reset circuit breakers and switches.

QUANTIFICATION

	Optimistic	Central	Pessimistic
Lfac:	Likely	Likely	Certain
L-AC:	Unlikely	Unlikely	Impossible

CASE #2:

This case is similar to Case 1 with the exception that the station batteries are not drained. The offsite power non-recovery probability is assessed directly from available data (NUREG-1032 data base) with no correction to account for loss of dc power.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LfAC:	Indeterminate	Likely	Likely
L-AC:	Indeterminate	Unlikely	Unlikely

CASE #3:

This case includes the so-called fast station blackout sequences in which there is no early high-pressure injection but dc power is available (TBU plant damage state). Vessel breach would occur approximately 3 hours following the loss of power. The probability of ac power recovery is taken to be the probability that it was recovered after 4.5 hours, given that it was not recovered after 3 hours.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LfAC:	Indeterminate	Likely	Likely
L-AC:	Indeterminate	Unlikely	Unlikely

CASE #4:

This case is similar to Case #3 with the exception that dc power is unavailable (TBUX plant damage state). As in Case 4 of Question #32 a correction is applied to the offsite power nonrecovery probability to account for the lack of dc power.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LfAC:	Likely	Likely	Certain
L-AC:	Unlikely	Unlikely	Impossible

CASE #5:

This case covers sequences in which ac power was not lost or was recovered prior to vessel breach. It is thus certain that ac power remains available.

QUANTIFICATION

	Optimistic	Central	Pessimistic
L+AC:	Impossible	Impossible	Impossible
L-AC:	Certain	Certain	Certain

QUESTION #85: Is RHR not operating late?

This question determines the availability of containment heat removal late in the sequence. Containment heat removal can be accomplished using either the spray or suppression pool cooling modes of RHR operation. Heat removal from the containment by venting or as a result of containment failure is considered in other questions.

This is a Type 2 question, and there are two branches:

1. nLRHR: There is no heat removal from the containment (using the RHR system) subsequent to vessel breach.
2. LRHR: The RHR system is used to remove heat from the containment subsequent to vessel breach.

CASE #1:

In this case, either the RHR system has previously failed, or there is no ac power available to operate the system, or the suppression pool is drained so there is no water to pump or the suppression pool is saturated (following containment failure) and thus the pumps are assumed not to have sufficient NPSH. Thus there is certainly no containment heat removal.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRHR:	Certain	Certain	Certain
LRHR:	Impossible	Impossible	Impossible

CASE #2:

In this case, the RHR was not previously available due to the lack of ac power. Power is restored after vessel breach. The RHR system cannot operate unless the pumps have sufficient Net Positive Suction Head (NPSH). Thus, either the containment must be intact or the suppression pool must be subcooled in order for RHR to be operable.

The RHR system is manually initiated. Quantification of the probability that the RHR system is operated for this case thus considers the probability that the operator will start the system. Operation is far from certain since the plant's condition has gone far beyond anything that plant personnel are trained to handle. The central estimate is that operation of RHR is very likely since this would be one of the few options available to the operator.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRHR:	Highly Unlikely	Very Unlikely	Indeterminate
LRHR:	Highly Likely	Very Likely	Indeterminate

CASE #3:

In this case, either the RHR has been operating until the time of vessel breach. Either the containment has remained intact or the suppression pool still contains subcooled water. Continued operation is certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRHR:	Certain	Certain	Certain
LRHR:	Impossible	Impossible	Impossible

QUESTION #86: Do containment sprays not operate following vessel breach?

This question determines whether the containment sprays are operating subsequent to vessel breach. Operation of the containment sprays will profoundly influence the release of fission products to the environment. In particular, the operation of sprays late in the sequence (during core-concrete interaction) will significantly reduce, and possibly prevent, the release of nonvolatile fission-product species. Only drywell spray provided by the RHR pumps is considered by this question.

This is a Type 2 question, and there are two branches:

1. nLCS: The containment (drywell) sprays are not operating subsequent to vessel breach.
2. LCS: The containment (drywell) sprays are operating subsequent to vessel breach.

CASE #1:

In this case, either the sprays failed earlier in the sequence (perhaps initially), or there is no ac power available to power the pumps or the suppression pool is saturated or drained (following containment failure). There is no late containment spray provided by the RHR pumps.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCS:	Certain	Certain	Certain
LCS:	Impossible	Impossible	Impossible

CASE #2:

In this case, sprays were operating prior to vessel breach. Either the containment has remained intact, or the suppression pool still contains subcooled water. Continued operation of the sprays is certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCS:	Impossible	Impossible	Impossible
LCS:	Certain	Certain	Certain

CASE #3:

In this case, the sprays were previously unavailable but ac power is recovered subsequent to vessel breach. In addition, either the containment is intact or the suppression pool is subcooled so if the operators align the RHR system for spray mode and start the pumps, there will be sufficient NPSH for the pumps to provide spray flow. (The suppression pool is not drained.) Quantification of this case thus reflects the probability that plant personnel will initiate sprays late in the sequence. Central estimate quantification is exactly as in Case #2 of the previous question, under the assumption that if sprays are operable, the operator would use them for RHR. If RHR cooling were unavailable (service-water failure) the operators would still use spray (with approximately the same probability) in an attempt to reduce containment pressure.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCS:	Highly Unlikely	Very Unlikely	Unlikely
LCS:	Highly Likely	Very Likely	Likely

QUESTION #87: Is service water sprayed following vessel breach?

This question considers whether the operator uses HPSW to provide spray to the drywell. For scenarios with ac power available, HPSW can provide spray when the RHR pumps cannot (because the suppression pool from which they draw suction is saturated or drained). This is a separate question because it is a key uncertainty in the LLH study, and sampling is facilitated by this structure (rather than combining this question with the previous one). It is noted that Peach Bottom procedures do not specifically instruct the use of HPSW for spray. However, this method for obtaining spray is recognized.

This is a Type 2 question, and there are two branches:

1. nLSWS: The operators do not use the HPSW system to provide spray to the drywell.

2. LSWS: Drywell spray is provided by the HPSW system.

CASE #1:

In this case, either HPSW is unavailable as a result of a system failure or because ac power is unavailable (the latter is the only contributor in the current assessment). If HPSW had previously been used for injection to the vessel, it is assumed that the operator would not divert the flow to spray since there would be no certain indication that the vessel had failed, and indeed it may not have. If the RHR system is being used for spray, it is assumed that the operators would certainly not consider using HPSW in its place.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSWS:	Certain	Certain	Certain
LSWS:	Impossible	Impossible	Impossible

CASE #2:

The case considers the probability that the operators will use the HPSW for spray. This is assumed to be the only source of late spray (spray using the RHR was considered in Question #86 and the previous case of this question. The probability is indeterminate since there are no specific procedures.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSWS:	Unlikely	Indeterminate	Impossible
LSWS:	Likely	Indeterminate	Certain

QUESTION #88: Is water not supplied to the debris late?

This question determines whether water is delivered to the core debris in the drywell subsequent to vessel breach. This is significant since water has the potential to quench the debris, halting the core-concrete interaction (CCI) and preventing the further release of fission products. Even if CCI is not prevented, fission-product releases will be mitigated.

This is a Type 2 question, and there are two branches:

1. nLDBWat: No water is supplied to the core debris on the drywell floor.
2. LDBWat: Water is supplied to the core debris.

CASE #1:

This case covers station blackout scenarios in which power is not recovered. Without power available, water cannot be supplied to the debris.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLDBWat:	Certain	Certain	Certain
LDBWat:	Impossible	Impossible	Impossible

CASE #2:

In this case, either the core debris is quenched in-vessel and there was no breach of the vessel, or water was being injected to the RPV at the time of RPV meltthrough. If meltthrough occurs, this water will continue to fall on the debris. This implies that water was supplied to the core debris.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLDBWat:	Impossible	Impossible	Impossible
LDBWat:	Certain	Certain	Certain

CASE #3:

In this case, the containment (drywell) sprays are operating. There is thus at least 5000 gpm of water being pumped into the drywell. This water will certainly reach the core debris.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLDBWat:	Impossible	Impossible	Impossible
LDBWat:	Certain	Certain	Certain

CASE #4:

In this case, there was water injection to the vessel (either condensate flow or HPSW) but the core debris was not quenched in-vessel. Injection to the vessel is assumed to continue and since the vessel is breached, this water is supplied to the debris on the drywell floor. This case is largely redundant to Case #2.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLDBWat:	Impossible	Impossible	Impossible
LDBWat:	Certain	Certain	Certain

CASE #5:

In this case, the condensate system is available to pump water into the vessel. Since the vessel bottom head is breached, if the condensate pumps were started, the water would reach the debris on the drywell floor. In the quantification of this case, it was considered that the condensate system is the system of choice for supplying water to the core. However, for these severely degraded conditions, it is not certain that the operators would avail themselves of this option. The central estimate of the failure probability is 0.01. A factor of 10 uncertainty is applied for the optimistic and pessimistic estimates.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLDBWat:	Highly Unlikely	Very Unlikely	Unlikely
LDBWat:	Highly Likely	Very Likely	Likely

CASE #6:

In this case, either the CRD flow was being injected to the vessel at the time of vessel breach. The probability of CRD flow failure resulting from containment rupture (should it occur) is judged to be zero for Peach Bottom, where the CRD pumps are in the turbine building. However, damage to the CRD hydraulic lines may occur at vessel breach which prevents further flow of water. For instance, the lines may melt and become embedded in solidified debris. It is considered likely that flow would continue.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLDBWat:	Very Unlikely	Unlikely	Indeterminate
LDBWat:	Very Likely	Likely	Indeterminate

CASE #7:

In this case, no pumps are operating prior to vessel breach because ac power has not been recovered. The CRD or HPSW pumps can be operated once power is restored. Quantification of this case reflects the probability that the operators will restart one, or both, of these systems following vessel breach. Owing to the confusion which is judged to be associated with power recovery, and given that the condensate system was not used, the assigned central estimate is less than certain success. This reflects complete uncertainty as to the actual value on the part of the analysts.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLDBWat:	Very Unlikely	Unlikely	Indeterminate
LDBWat:	Very Likely	Likely	Indeterminate

CASE #8:

This case covers sequences in which there are no systems available for the injection of water into either the vessel or the drywell. Therefore, no water can be supplied to the core debris. This case is not utilized for plant damage states in the current ASEP profile.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLDBWat:	Certain	Certain	Certain
LDBWat:	Impossible	Impossible	Impossible

QUESTION #89: Is the debris not coolable without water?

This question determines whether the core debris, which has fallen or been ejected from the vessel, is distributed such that it gives up sufficient heat to the drywell to prevent core-concrete interaction from being sustained. The heat loss mechanisms would be direct radiation to the drywell structures, convection, and conduction to the surfaces on which the debris has been deposited.

This is a Type 2 question, and there are two branches:

1. nDBCwoW: The core debris cannot be cooled without water.

2. DBCwoW: The debris is distributed in the drywell such that it is self-cooling (no water needed).

CASE #1:

In this case, debris has been dispersed in the drywell either as a result of a steam explosion (in-vessel or ex-vessel) or direct heating has occurred. If there has been direct heating, the debris will necessarily have been dispersed. Dispersal does not assure coolability, but with any of these events, the dispersal is expected to be widespread within the drywell such that there is a probability that core-concrete interaction cannot be sustained. The central estimate is that coolability without water is highly unlikely (0.0001 probability). Pessimistically, it is impossible. The optimistic estimate allows a 0.1 probability so that the possibility is recognized.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nDBCwoW:	Likely	Highly Likely	Certain
DBCwoW:	Unlikely	Highly Unlikely	Impossible

CASE #2:

This case covers sequences in which there has been no energetic dispersal of the core debris. The debris is thus pooled on the drywell floor. In this geometry, cooling by convection and radiant transfer to the drywell structures is not expected to be adequate to remove the decay heat plus the heat generated as a result of reactions in the melt (i.e., not coolable without water).

QUANTIFICATION

	Optimistic	Central	Pessimistic
nDBCwoW:	Certain	Certain	Certain
DBCwoW:	Impossible	Impossible	Impossible

QUESTION #90: Is the debris coolable with water?

This question determines whether not the core debris can be cooled with water. There is phenomenological uncertainty with regard to whether the heat transfer to water covering a pool of molten corium that is reacting with concrete is adequate to remove the heat generated within the melt. Recent experiments performed at SNL indicate that the addition of water to the melt will not prevent core concrete interaction from continuing. However, analyses performed by IDCOR indicate that the addition of water will result in quenching of the melt.

This is a Type 2 question, and there are two branches:

1. nDBC: The core debris in the drywell is not coolable.
2. DBCwW: The core debris in the drywell can be quenched (further core-concrete interaction prevented) by supplying water to it.

CASE #1:

In this case, either the debris was dispersed sufficiently at vessel breach such that it can be cooled even without water (see the previous question) or the RPV was not breached. If the latter is true, it is only because the debris was coolable with water that the vessel was not breached. Thus, in either instance, the debris can certainly be cooled by supplying water to it.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nDBC:	Impossible	Impossible	Impossible
DBCwW:	Certain	Certain	Certain

CASE #2:

In this case, as in Case #1 of the previous question, the debris was energetically dispersed at the time of vessel breach. Sequences in which this dispersal was sufficient to allow cooling without water were considered under the previous case of this question. However, given that the debris cannot be cooled without water, there is still a significant probability that it can be cooled by the addition of water. A judgment of this probability is reflected by the quantification of this case. There is no analysis for this case so the central estimate probability of failure is 0.5. The optimistic and pessimistic estimates reflect a reasonable level of uncertainty in the quantification.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nDBC:	Unlikely	Indeterminate	Likely
DBCwW:	Likely	Indeterminate	Unlikely

CASE #3:

In this case, there was a slump-type melt in-vessel. The core debris was dumped on the pedestal floor en masse when the vessel was breached. A significant fraction of the debris is expected to be molten; thus, there is expected to be some flowing of the melt out from the pedestal area (through the manway) onto the drywell floor. With the debris in one lump and substantial fraction being molten, the probability of it being coolable is judged to be at a minimum. On the basis of discussions with D. Powers of SNL and G. Green at BNL, who both ascribe little credibility to corium being quenched by an overlaying pool of water, the probability of quenching is set to 0.1 in the central estimate. The optimistic estimate is 0.9, based on the IDCOR analysis, which shows that cooling would occur. (This analysis has not been reviewed by the authors). The pessimistic estimate is that there is a negligible probability of coolability.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nDBC:	Unlikely	Likely	Certain
DBCwW:	Likely	Unlikely	Impossible

CASE #4:

In this case, there was injection to the RPV at the time of vessel breach. Thus, water and core debris flow from the hole in the vessel. Since this case considers only flow-type melt progressions (described under Question #68) a substantial portion of the debris is solid. Water mixes with the debris prior to the debris falling on the drywell floor. This is judged to slightly enhance the probability of debris coolability (compared to the succeeding case). (An input error for this case resulted in the probability assignments for the two branches being reversed for all three estimates. A slight impact on the results is anticipated.)

QUANTIFICATION

	Optimistic	Central	Pessimistic
nDBC:	Unlikely	Likely	Likely
DBCwW:	Likely	Unlikely	Unlikely

CASE #5:

This case covers sequences in which there was a flow-type melt. Melted debris leaving the vessel is expected to be closer to the solidus temperature than for a slump-type of in-vessel melt. For this type of melt, there is a greater chance that the debris will (at least initially) be retained in the pedestal area. However, whether the geometry of the debris resulting from this type of core melt is more or less easily cooled is not known. Since the debris comes out gradually at lower temperature, there is reason to believe that a coolable geometry will be formed (initially the debris falls into water present in the drywell sumps). With no analysis available, the probability that the debris will not be coolable is 0.5. Again, the optimistic and pessimistic estimates reflect a reasonable range of uncertainty.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nDBC:	Impossible	Indeterminate	Likely
DBCwW:	Certain	Indeterminate	Unlikely

QUESTION #91: Is the debris not cooled?

This question summarizes the results of the three previous questions with respect to whether core-concrete interaction (CCI) is prevented or mitigated by cooling the core debris.

This is a Type 2 question, and there are two branches:

1. nCoolDB: The core debris in the drywell is not cooled sufficiently to remove the decay heat plus the heat generated by core-concrete interaction.
2. CoolDB: The core debris in the drywell is cooled sufficiently to prevent further core concrete interaction.

CASE #1:

In this case, either the debris received adequate cooling in-vessel such that vessel breach was prevented or the debris was sufficiently dispersed at vessel breach that cooling is assured even without water. Thus, the debris is certainly cooled.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nCoolDB:	Impossible	Impossible	Impossible
CoolDB:	Certain	Certain	Certain

CASE #2:

In this case, the debris in the drywell can be cooled if water is supplied to it and some method of delivering water flow to the drywell has been established. On the basis of the way the previous questions are defined, the debris is certainly cooled.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nCoolDB:	Impossible	Impossible	Impossible
CoolDB:	Certain	Certain	Certain

CASE #3:

This case covers sequences in which either the debris is not coolable or there is no water supplied to it. Either way, the debris is not cooled and core concrete interaction will proceed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nCoolDB:	Certain	Certain	Certain
CoolDB:	Impossible	Impossible	Impossible

QUESTION #92: Is the containment not vented late?

This question determines whether the containment is vented from the wetwell airspace after vessel breach.

This is a Type 2 question, and there are two branches:

1. LnVent: The operator does not (or is not able to) vent the containment after vessel breach. (Note: The containment may have been previously vented.)

2. L-Vent: The plant operators successfully vent the containment only after the vessel has been breached.

CASE #1:

In this case either the containment has already lost its integrity and the pressure can no longer reach the level at which the procedures require venting or the RPV has not been breached. Containment pressure is no longer rising in either case. Late venting is thus assigned a negligible probability.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LnVent:	Certain	Certain	Certain
L-Vent:	Impossible	Impossible	Impossible

CASE #2:

This case covers blackout sequences in which ac power is not recovered. The containment pressure does not exceed the threshold for venting prior to breach of the RPV. Question #47 considers cases where containment pressure exceeded the venting threshold prior to RPV breach. It is judged that if the operators did not vent when pressure first exceeded the threshold, they would not do so at some later time. This case is quantified identically with Case #7 of Question #47; from the operator's perspective, the situation is similar.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LnVent:	Indeterminate	Likely	Certain
L-Vent:	Indeterminate	Unlikely	Impossible

CASE #3:

In this case, the accident has essentially been arrested. Decay heat is being removed from the containment via the RHR heat exchangers and core-concrete interaction has been terminated. Since the containment pressure would be decreasing, there is no necessity for venting.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LnVent:	Certain	Certain	Certain
L-Vent:	Impossible	Impossible	Impossible

CASE #4:

In this case, either ac power is first recovered subsequent to vessel breach, or containment pressure first exceeded the venting threshold subsequent to vessel breach. In both situations, ac power is available and the containment pressure continues to rise. This case is quantified identically with Case #5 of Question $47; from the operator's perspective, the situation is similar.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LnVent:	Highly Unlikely	Very Unlikely	Unlikely
L-Vent:	Highly Likely	Very Likely	Likely

CASE #5:

In this case, the venting-threshold pressure was exceeded during core damage. The operators failed to vent at that time and their ability to do so has not changed. A negligible probability is assigned to belated venting.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LnVent:	Certain	Certain	Certain
L-Vent:	Impossible	Impossible	Impossible

QUESTION #93: Does the containment fail late due to temperature?

This question determines whether there is a failure of drywell penetration seals as a result of elevated temperature. High-temperature failures of the drywell penetration seals was the assumed mode of failure in the IDCOR Task 23.1 report on Peach Bottom. Loss of drywell integrity due to high temperatures can occur even if the containment has previously failed in the wetwell or been vented. Thus, such failures may significantly impact fission product release to the environment.

This is a Type 2 question, and there are two branches:

1. LTCF: There is a failure of one or more drywell penetration seals, due to elevated temperatures in the drywell, subsequent to vessel breach.

2. nLTCF: There are no drywell penetration seal failures resulting from high drywell temperatures.

CASE #1:

This case considers accident progressions in which either there is no severe thermal loading to the drywell or the drywell has already failed. In the latter situation, thermal failure of the penetration seals is irrelevant. If the vessel has not been breached, or if water is being supplied to core debris on the drywell floor, penetration seals will not see temperatures high enough to result in failure.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LTCF:	Impossible	Impossible	Impossible
nLTCF:	Certain	Certain	Certain

CASE #2:

In this case, the core debris is distributed in the drywell and is giving up its heat (decay power) to the drywell structures and equipment. No water is being supplied to the drywell so there is no effective mechanism for removing this heat. The drywell temperature will continue to rise until heat conduction through the containment wall reaches equilibrium with the decay heat being generated. This would not occur until the temperature in the drywell, and in the walls, was well above what the seals could withstand. Temperature-induced failure is thus certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LTCF:	Certain	Certain	Certain
nLTCF:	Impossible	Impossible	Impossible

CASE #3:

In this case, containment pressure is low due to venting or previous containment failure. While the inner surface of the penetration seals would be exposed to high temperatures,

314

the outer surface would take much longer to heat up as a result of conduction. Since the containment is depressurized, there is also little pressure on the seals that could drive them to leak. Thus, the failure probability in the central estimate is 0.1. The optimistic estimate is a factor of 10 less. In the pessimistic estimate, the evaluation is 0.5; reflecting the lack of analysis.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LTCF:	Very Unlikely	Unlikely	Indeterminate
nLTCF:	Very Likely	Likely	Indeterminate

CASE #4:

This case covers accident progressions in which there is debris on the drywell floor and core-concrete interaction is occurring. Heat generated in the melt is either transferred to the concrete, carried away by the gases being generated (most of this portion will be absorbed by the pool and/or released from the containment), or goes into the drywell equipment and structures. The magnitude of this latter fraction (and also the extent of exothermic reactions in the melt) largely determines the drywell temperature. The IDCOR Task 23.1 assessment was that the drywell temperature would rise sufficiently to cause loss of the seal integrity. However, there is sufficient uncertainty in the CCI phenomena that such a consequence is not certain. The central estimate of the thermal failure probability is 0.5. The optimistic and central estimates reflect reasonable uncertainty in this value.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LTCF:	Unlikely	Indeterminate	Likely
nLTCF:	Likely	Indeterminate	Unlikely

QUESTION #94: Does the containment fail late?

This question determines whether containment integrity is lost as a result of pressurization due to gases generated during core-concrete interaction, or containment heat-up, after vessel breach.

This is a Type 2 question, and there are three branches:

1. LPCF: The containment fails as a result of overpressurization due to core-concrete interaction (or the containment is vented).
2. LTCF: The containment fails as a result of drywell penetration seal failure after vessel breach.
3. nLCF: Containment integrity is maintained throughout the accident.

CASE #1:

In this case, the drywell penetration seals have failed. Containment integrity is thus lost as a result of high temperatures.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LPCF:	Impossible	Impossible	Impossible
LTCF:	Certain	Certain	Certain
nLCF:	Impossible	Impossible	Impossible

CASE #2:

In this case, water is being supplied to the core debris on the drywell floor, which is thereby successfully cooled. The containment heat removal (or RHR) system is operating. The heat generated by core debris in the drywell will be transferred to the suppression pool by the water flow through the drywell vents back into the suppression pool. Under these conditions, the containment pressure will be decreasing, and the containment integrity will not be further threatened.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LPCF:	Impossible	Impossible	Impossible
LTCF:	Impossible	Impossible	Impossible
nLCF:	Certain	Certain	Certain

CASE #3:

In this case, there is water being supplied to the core debris on the drywell floor and the debris is thereby successfully cooled. However, there is no heat removal from the containment and the containment has not been vented, so

316

the containment will continue to pressurize. Eventual overpressure failure is thus certain. (Note: Sequences with earlier containment failure are excluded from this case but are reflected in subsequent questions with regard to leakage.)

QUANTIFICATION

	Optimistic	Central	Pessimistic
LPCF:	Certain	Certain	Certain
LTCF:	Impossible	Impossible	Impossible
nLCF:	Impossible	Impossible	Impossible

CASE #4:

In this case, the debris is not cooled but containment heat removal is operating which implies that water covers the debris, keeping drywell temperatures moderate. The containment will fail as a result of the overpressure from noncondensable gases (carbon dioxide, hydrogen, carbon monoxide) produced by CCI.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LPCF:	Certain	Certain	Certain
LTCF:	Impossible	Impossible	Impossible
nLCF:	Impossible	Impossible	Impossible

CASE #5:

In this case, water is being supplied to the core debris on the drywell floor, but core-concrete interaction continues. The containment is still intact after vessel breach, but pressurization continues. The presence of water in the drywell will keep drywell temperatures below levels at which seal failures due to temperature are a significant concern. Containment failure as a result of overpressurization by the combination of steam and/or noncondensable gases is certain. This case is largely redundant to the previous one. However, in Case #4, RHR operation reduces the steam partial pressure, delaying containment failure relative to this case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LPCF:	Certain	Certain	Certain
LTCF:	Impossible	Impossible	Impossible
nLCF:	Impossible	Impossible	Impossible

CASE #6:

In this case, no water is supplied to the core debris in the drywell and core-concrete interaction continues. The containment is intact subsequent to vessel breach, but pressurization continues due to the heat and noncondensable gas generation. Drywell thermal failures were previously considered (Question #93 and Case #1 of this question), so the only type of failure considered here is failure due to overpressure, which is certain to occur.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LPCF:	Certain	Certain	Certain
LTCF:	Impossible	Impossible	Impossible
nLCF:	Impossible	Impossible	Impossible

CASE #7:

This case covers sequences in which the containment has previously been failed or vented. The containment pressure is not rising, and secondary failure of the drywell penetration seals due to high drywell temperatures has be previously considered. (Question #93 and Case #1 of this question.) No late failure of containment will occur.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LPCF:	Impossible	Impossible	Impossible
LTCF:	Impossible	Impossible	Impossible
nLCF:	Certain	Certain	Certain

QUESTION #95: What is the location of late containment leakage?

This question determines the location of containment leakage. The leakage location profoundly influences fission product retention along the release pathway to the environment. This question is analogous to previous questions (Question #28, Question #51, and Question #80).

This is a Type 2 question, and there are three branches:

1. nLCL: There is no leakage from the containment in excess of technical specification limits.

2. LCLD: Containment leakage subsequent to vessel breach is from the drywell.

3. LCLW: Containment leakage subsequent to vessel breach is from the wetwell (torus).

CASE #1:

In this case, there has been previous leakage from the drywell. Accident progressions in which previous drywell leakage was Level 2 and venting occurred late are excluded from this case. Continued leakage from the drywell is certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCLD:	Certain	Certain	Certain
LCLW:	Impossible	Impossible	Impossible

CASE #2:

In this case, drywell failure occurred as a result of elevated temperatures (determined by Question #93). Accident progressions in which the torus previously ruptured or was vented are excluded from this case if the suppression pool is bypassed. These are excluded since, with the drywell vents uncovered, the pathway through the wetwell offers less resistance to flow than the pathway opened by thermal failure. If the suppression pool were not bypassed, the static head of water covering the vent downcomers would significantly increase flow resistance through the wetwell pathway.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCLD:	Certain	Certain	Certain
LCLW:	Impossible	Impossible	Impossible

CASE #3:

In this case, either the wetwell has been successfully vented or previously developed leakage. Excluding the case of thermal failures (previous case of this question), the dominant leakage path will continue to be through the wetwell.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCLD:	Impossible	Impossible	Impossible
LCLW:	Certain	Certain	Certain

CASE #4:

This case includes sequences in which the containment fails as a result of overpressurization but in which the drywell temperatures are maintained below approximately 500°F by the presence of water. For such sequences (drywell temperature moderate) the split fraction between wetwell and drywell failure location is determined exactly as for sequences in which containment failure occurs prior to vessel breach (Case #4 of Question #28).

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCLD:	Indeterminate	Likely	Certain
LCLW:	Indeterminate	Unlikely	Impossible

CASE #5:

This case is similar to the previous one in that there is an overpressurization failure of the containment late in the sequence. However, in this case, water is not supplied to the debris, and the drywell temperature is expected to be quite high (greater than 500°F). The central estimate of the split fraction between wetwell and drywell failure is based on the opinions of the containment performance experts who were convened to review uncertainty issues for SARRP [6]. The optimistic and pessimistic quantifications of the split fraction reflect the range in their opinions.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCLD:	Indeterminate	Very Likely	Certain
LCLW:	Indeterminate	Very Unlikely	Impossible

CASE #6:

This case covers sequences in which the containment remained intact throughout the accident. There is thus negligible probability of leakage from either the drywell or the wetwell.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Certain	Certain	Certain
LCLD:	Impossible	Impossible	Impossible
LCLW:	Impossible	Impossible	Impossible

QUESTION #96: For drywell failures, is the failure at the head seal?

This question is completely analogous to Question #29.

This is a Type 2 question, and there are three branches:

1. LDHF: The dominant leakage path from the containment late in the accident is from the drywell head seal.

2. LDKF: The dominant leakage path from the containment late in the accident is from the drywell, below the head seal (knuckle region or below) to the reactor building.

3. nLDF: The dominant leakage path from the containment (if any) is not from the drywell.

CASE #1:

In this case, high temperatures and pressures in the drywell have led to thermal failure of the penetration seals. These seals are in the drywell wall below the head closure. Since leakage flow from these seals will generally go to the reactor building, this failure mode is considered equivalent to a failure at the drywell knuckle.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LDHF:	Impossible	Impossible	Impossible
LDKF:	Certain	Certain	Certain
nLDF:	Impossible	Impossible	Impossible

CASE #2:

In this case, there was prior leakage from the containment from the drywell head seal. The drywell penetration seals remain intact and the drywell wall has not melted-through. It is assumed that the dominant leakage pathway will continue to be from the drywell head seal.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LDHF:	Certain	Certain	Certain
LDKF:	Impossible	Impossible	Impossible
nLDF:	Impossible	Impossible	Impossible

CASE #3:

In this case, there was previous drywell leakage from the knuckle (or equivalent) region. It is judged that this is certain to remain the dominant leakage pathway.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LDHF:	Impossible	Impossible	Impossible
LDKF:	Certain	Certain	Certain
nLDF:	Impossible	Impossible	Impossible

CASE #4:

In this case, overpressurization failure occurred in the drywell late in the accident sequence. The situation considered is the same as for Case #4 of Question #95; that is, delivery of water to the drywell is keeping temperatures at moderate (<500°F) levels. The split fraction between head seal and knuckle failure is thus expected to be the same as it would be earlier in the accident. The split fraction between the head seal and knuckle (or equivalent) locations is determined on the basis of the expert responses. The quantification is identical to Case #3 of Question #29.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LDHF:	Unlikely	Indeterminate	Likely
LDKF:	Likely	Indeterminate	Unlikely
nLDF:	Impossible	Impossible	Impossible

CASE #5:

Like the previous case, this case considers the split fraction between head seal and knuckle region failure, given overpressure failure of the drywell. However, in this case, no water is supplied to the core debris on the drywell floor. Drywell temperatures at the time of failure are thus expected to be significantly higher. The SARRP containment performance expert reviewers indicated a higher probability for failure at the head seal should be applied for elevated drywell temperatures [6]. This judgment is reflected in the quantification of this case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LDHF:	Unlikely	Indeterminate	Likely
LDKF:	Likely	Indeterminate	Unlikely
nLDF:	Impossible	Impossible	Impossible

CASE #6:

In this case, the drywell remains intact throughout the accident. There is thus no leakage from the drywell.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LDHF:	Impossible	Impossible	Impossible
LDKF:	Impossible	Impossible	Impossible
nLDF:	Certain	Certain	Certain

QUESTION #97: What is the level of late containment leakage?

This question determines the level of leakage from the containment in the time period after the beginning of core-concrete interaction or the time of containment failure (whichever is earlier).

This is a Type 2 question, and there are three branches:

1. nLCL: The containment retains its integrity over the entire course of the accident. Leakage rates are not significantly higher than technical specification limits.

2. LCL2: The containment develops a leak, but the leak rate is sufficient to cause only slow (>2 hours) depressurization of the containment.

3. LCL3: The containment is ruptured, and the interior pressure is approximately atmospheric pressure.

CASE #1:

In this case, the containment was previously ruptured and thus remains ruptured.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCL2:	Impossible	Impossible	Impossible
LCL3:	Certain	Certain	Certain

CASE #2:

This case includes late thermal failures in the drywell. The IDCOR position is that this failure mode is certain (or at least highly likely--the optimistic assumption here) to result only in a leak (Level 2). However, once leakage began through the seals, there would seem to be a significant probability that the seals would erode; resulting in a leakage area equivalent to a rupture. Since there has been no analysis of this failure mode, a probability of 0.5 is assigned to the scenario that leakage equivalent to a rupture developed. The pessimistic case assumes rupture to be certain.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCL2:	Highly Likely	Indeterminate	Impossible
LCL3:	Highly Unlikely	Indeterminate	Certain

CASE #3:

In this case, overpressurization failure of the containment occurred at the head seal. Consistent with the previous questions with regard to containment leakage level (Question #30 and Question #81), this failure location is always assumed to result in Level 2 leakage.

324

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCL2:	Certain	Certain	Certain
LCL3:	Impossible	Impossible	Impossible

CASE #4:

This case considers accident progressions in which the location of late containment failure due to overpressure is at moderate temperature. Thus, scenarios with wetwell failure and drywell failure with concurrent delivery of water to the core debris are included. The consensus opinion of the containment performance experts who considered this question for SARRP was that the split between leak and rupture-type failures was 50-50 [6]. No analysis is available that provides other information.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCL2:	Likely	Indeterminate	Unlikely
LCL3:	Unlikely	Indeterminate	Likely

CASE #5:

This case is the complement of the previous one in the consideration of late containment failures. Scenarios with overpressure in the drywell with no water supplied to the core debris are included. In the judgment of the analysts, the elevated temperature has the potential to impact the induced leakage area. For instance, a creep-induced failure, which would tend to have lower leakage area, may be more likely. However, the SARRP experts assigned a similar average split fraction for this case [6]. Lacking evidence to support some other quantification, the quantification of branch point probabilities for this case are identical to the previous case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCL2:	Likely	Indeterminate	Unlikely
LCL3:	Unlikely	Indeterminate	Likely

CASE #6:

In this case, the containment has been vented late in the sequence. Venting will result in containment depressurization and is thus a Level 3 leak, defined to be equivalent to a rupture of the containment.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCL2:	Impossible	Impossible	Impossible
LCL3:	Certain	Certain	Certain

CASE #7:

In this case, the containment was leaking (Level 2) immediately after vessel breach and subsequent progression of the accident has not placed additional loading on the containment. Leakage is considered to continue and the containment pressure will continue to gradually decline.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Impossible	Impossible	Impossible
LCL2:	Certain	Certain	Certain
LCL3:	Impossible	Impossible	Impossible

CASE #8:

This case includes sequences in which the accident is recovered either prior to or shortly after vessel breach and the containment had not previously failed. Since there is recovery, by definition, the containment integrity is maintained. The containment leakage rate is not significantly higher than technical specification limits.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLCL:	Certain	Certain	Certain
LCL2:	Impossible	Impossible	Impossible
LCL3:	Impossible	Impossible	Impossible

QUESTION #98: Is the suppression pool drained late?

This question determines whether the suppression pool still contains enough water to cover the drywell vents and therefore maintain fission product scrubbing for sequences in which the dominant release pathway is through the wetwell. If the containment has failed in the wetwell, this allows the possibility that the failure has occurred below the pool water level. Partial (or complete) draining of the pool could then occur. The level need only be lowered 3 to 4 feet before there will be essentially no scrubbing of the gas effluent from the containment.

This is a Type 2 question, and there are three branches:

1. LSPD: The suppression pool is drained. Both the drywell vents and the ECC suction lines are uncovered.

2. pLSPD: The suppression pool level is reduced below the level of the drywell vents. Releases from the drywell bypass the suppression pool.

3. nLSPD: The suppression pool is not drained. The drywell vents remain submerged in the suppression pool.

CASE #1:

In this case, the pool level was previously drained. The pool level will remain too low, even for operation of the ECC pumps.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LSPD:	Certain	Certain	Certain
pLSPD:	Impossible	Impossible	Impossible
nLSPD:	Impossible	Impossible	Impossible

CASE #2:

In this case, the suppression pool level has previously been reduced below the level of the drywell vents. Further reduction or increase in the level is considered impossible.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LSPD:	Impossible	Impossible	Impossible
pLSPD:	Certain	Certain	Certain
nLSPD:	Impossible	Impossible	Impossible

CASE #3:

In this case, the torus has ruptured late in the accident. Quantification of this case is identical to Case #1 of Question #31.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LSPD:	Impossible	Very Unlikely	Unlikely
pLSPD:	Very Unlikely	Indeterminate	Likely
nLSPD:	Very Likely	Indeterminate	Unlikely

CASE #4:

This case covers both sequences in which containment failure occurred prior to vessel breach in which pool level was maintained and sequences with later failures in the drywell. The pool level is not reduced in either situation.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LSPD:	Impossible	Impossible	Impossible
pLSPD:	Impossible	Impossible	Impossible
nLSPD:	Certain	Certain	Certain

QUESTION #99: What is the level of late suppression pool bypass?

This question determines the extent to which the suppression pool is bypassed. Suppression pool bypass can profoundly influence the quantity of fission products released to the reactor building since pool scrubbing will not be effective.

This is a Type 2 question, and there are three branches:

1. nLSPB: The dominant release pathway for fission products to the reactor building is through the suppression pool.
2. LSPB2: There is a partial bypass of the suppression pool. A significant fraction of the gas flow from the containment is not scrubbed.
3. LSPB3: Gas flow from the containment completely bypasses the suppression pool. There is no fission product scrubbing.

328

CASE #1:

In this case, the vessel has been breached and either the dominant release pathway from the containment is through the drywell or the suppression pool has been drained or otherwise bypassed. Suppression pool bypass is complete.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSPB:	Impossible	Impossible	Impossible
LSPB2:	Impossible	Impossible	Impossible
LSPB3:	Certain	Certain	Certain

CASE #2:

In this case, there has been an induced primary system break in the reactor building. The suppression pool is thus certainly bypassed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSPB:	Impossible	Impossible	Impossible
LSPB2:	Impossible	Impossible	Impossible
LSPB3:	Certain	Certain	Certain

CASE #3:

This case considers accident progressions in which the vessel is not breached but the drywell has failed. In addition, there is a stuck-open SRV tailpipe vacuum breaker. Complete bypass of the suppression pool results.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSPB:	Impossible	Impossible	Impossible
LSPB2:	Impossible	Impossible	Impossible
LSPB3:	Certain	Certain	Certain

CASE #4:

In this case, the vessel was not breached. However, there was a previous partial bypass (Level 2) of the suppression pool. This bypass remains.

329

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSPB:	Impossible	Impossible	Impossible
LSPB2:	Certain	Certain	Certain
LSPB3:	Impossible	Impossible	Impossible

CASE #5:

In this case, the RPV has not been breached. However, drywell failure has occurred and an SRV tailpipe vacuum breaker failed to reseat. Partial bypass of the suppression pool results.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSPB:	Impossible	Impossible	Impossible
LSPB2:	Certain	Certain	Certain
LSPB3:	Impossible	Impossible	Impossible

CASE #6:

This case covers sequences in which the containment either, remains intact, has been vented, or in which failure of the containment occurs in the wetwell above the suppression pool water level. Pool scrubbing will retain a significant quantity of fission products in the pool during these sequences; there is no pool bypass.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSPB:	Certain	Certain	Certain
LSPB2:	Impossible	Impossible	Impossible
LSPB3:	Impossible	Impossible	Impossible

QUESTION #100: Do drywell sprays continue?

This question considers whether or not containment spray (if it has been previously activated) will continue until the end of the accident.

This is a Type 2 question, and there are two branches:

1. nVLDS: Drywell spray is not available following late containment failure.

2. VLDS: Drywell spray continues to the termination of the accident.

CASE #1:

This case considers accident progressions in which there was no drywell spray following vessel breach (after consideration of recovery). There is thus no spray following that time.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nVLDS:	Certain	Certain	Certain
VLDS:	Impossible	Impossible	Impossible

CASE #2:

In this case, late containment spray was provided by the RHR pumps. HPSW is failed or is providing injection to the RPV, and will not be used for drywell spray. Late containment failure causes drywell sprays to stop. This results either because the suppression pool saturates or because the suppression pool drains. Drywell spray is terminated.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nVLDS:	Certain	Certain	Certain
VLDS:	Impossible	Impossible	Impossible

CASE #3:

This case is similar to the preceding one except that HPSW is available to provide drywell spray after the RHR pumps stop. This case is quantified identically with Case 2 of Question #87, which considers similar circumstances.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nVLDS:	Unlikely	Indeterminate	Impossible
VLDS:	Likely	Indeterminate	Certain

CASE #4:

This case covers accident progressions in which drywell spray has been operating and nothing has occurred to interfere with that operation.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nVLDS:	Impossible	Impossible	Impossible
VLDS:	Certain	Certain	Certain

QUESTION #101: What is the level of late RB bypass without a burn?

This question determines the extent to which fission products released from the containment subsequent to vessel breach could be retained in the reactor building. There is an implicit assumption that fission product retention in the building is proportional to the gas residence time. The extent to which the gas bypasses the building (the length of the release pathway) and the flowrate through the building are thus considered. This basic assumption may be in conflict with the IDCOR model, which appears to rely heavily on impaction for fission product retention in the reactor building. This question is completely analogous to Question #59 above. More complete documentation is given thereunder.

This is a Type 2 question, and there are four branches:

1. nLRBB: There is no breach or bypass of the reactor building. What leakage there may be from the containment could be handled by the Standby Gas Treatment System (SGTS), assuming that it were operational.

2. LRBB2: There is leakage from the reactor building but all of that flow passes through the building on the pathway from the primary containment.

3. LRBB3: The reactor building is essentially bypassed. Either primary containment leakage is directly through the refueling bay or there is sufficient damage to the reactor building that retention along the release pathway would be similar to that case.

4. LRBB4: The reactor building is completely bypassed. There would be no significant fission product retention.

CASE #1:

In this case, vessel breach was the result of an Alpha mode steam explosion. By definition, all containment (including the reactor building) is completely bypassed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Impossible	Impossible	Impossible
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Certain	Certain	Certain

CASE #2:

This case considers accident progressions in which meltthrough of the drywell wall occurs subsequent to vessel breach. The reactor building has not been structurally damaged as a result of hydrogen burns. Leakage from the containment through the hole induced by meltthrough is assumed to go to the torus room in the reactor building basement. Since the reactor building is intact except for leakage through blowout panels, Level 2 bypass is assumed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Certain	Certain	Certain
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Impossible	Impossible	Impossible

CASE #3:

In this case, the reactor building was bypassed prior to vessel breach. The building will thus remain bypassed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Impossible	Impossible	Impossible
LRBB3:	Certain	Certain	Certain
LRBB4:	Impossible	Impossible	Impossible

CASE #4:

In this case, the reactor building was previously essentially bypassed (Level 3); it will remain so.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Impossible	Impossible	Impossible
LRBB3:	Certain	Certain	Certain
LRBB4:	Impossible	Impossible	Impossible

CASE #5:

This case is identical to Case #3 of Question #59. Leakage from the drywell is from the head seal to the refueling bay. It is quantified in the same manner.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Impossible	Impossible	Impossible
LRBB3:	Certain	Certain	Certain
LRBB4:	Impossible	Impossible	Impossible

CASE #6:

In this case, the primary containment has ruptured either in the wetwell or in the drywell below the head seal. As in Case #5 of Question #59, this is defined to result in a Level 2 bypass of the reactor building. This case also includes sequences in which the reactor building previously has a Level 2 bypass. This is assumed certain to remain at this level in the absence of further hydrogen burns (considered in subsequent questions).

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Certain	Certain	Certain
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Impossible	Impossible	Impossible

CASE #7:

In this case, the primary containment has developed Level 2 leakage (defined in Question #30). This case is identical to Case #6 of Question #59.

332

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Likely	Indeterminate	Unlikely
LRBB2:	Unlikely	Indeterminate	Likely
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Impossible	Impossible	Impossible

CASE #8:

This case covers all sequences in which the primary containment remains intact. With the primary containment intact, the reactor building will remain intact.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Certain	Certain	Certain
LRBB2:	Impossible	Impossible	Impossible
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Impossible	Impossible	Impossible

QUESTION #102: Are fire systems operating late without a late HB?

This question determines whether the fire sprays in the reactor building are running at the time fission products that are released subsequent to vessel breach enter. Operation of the fire sprays at some plants can increase fission product retention in the reactor building. However, because the Peach Bottom fire sprays are only a curtain spray between the two units, the increased retention will not be significant. The fire spray system is described above under the discussion of Question #60.

This is a Type 2 question, and there are two branches:

1. LFireS: The reactor building fire sprays are running subsequent to vessel breach.
2. nLFireS: The reactor building fire sprays are not started as a result of a late failure of the containment.

CASE #1:

In this case, the fire spray system was actuated prior to vessel breach. The fire sprays will continue to run.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Certain	Certain	Certain
nLFireS:	Impossible	Impossible	Impossible

CASE #2:

In this case, the accident is arrested prior to containment failure. There is no release to the reactor building, and thus the fire spray system would not be actuated.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Impossible	Impossible	Impossible
nLFireS:	Certain	Certain	Certain

CASE #3:

In this case, the reactor building is completely bypassed. With no accumulation of hot steam or gases in the areas of the reactor building that house the cable trays, the fire sprays would not be triggered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Impossible	Impossible	Impossible
nLFireS:	Certain	Certain	Certain

CASE #4:

In this case, the containment failure location is at the drywell head seal. Leakage from the head closure would pass up into the refueling area. Leakage of hot gas into this region is assumed not to result in the fire sprays being triggered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Impossible	Impossible	Impossible
nLFireS:	Certain	Certain	Certain

CASE #5:

In this case, the release pathway from the containment to the reactor building is through a subcooled suppression pool. With only warm gases being passed into the reactor building, the fire sprays would not be triggered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Impossible	Impossible	Impossible
nLFireS:	Certain	Certain	Certain

CASE #6:

This case covers all sequences in which there is release of hot gas from the containment to the reactor building. The building is not completely bypassed and so there will be significant heating of the building. This heating will cause the fire sprays to be triggered.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Certain	Certain	Certain
nLFireS:	Impossible	Impossible	Impossible

QUESTION #103: Does standby gas treatment not work late without late HB?

This question determines whether the Standby Gas Treatment System (SGTS) is operating during the period in which fission products, released from the containment after vessel breach, are in the reactor building. The impact of possible burning of combustible gases in the reactor building on SGTS operation is considered in a subsequent question. SGTS characteristics have been described previously (Question #61).

This is a Type 2 question, and there are two branches:

1. nLSGTS: The Standby Gas Treatment System does not operate subsequent to vessel breach.
2. LSGTS: The Standby Gas Treatment System is operating after the containment has failed, following vessel breach.

CASE #1:

In this case, the SGTS has not previously failed, and there is ac power available to operate it. The fire dampers in the SGTS duct work are still held open by the fusible links. Since SGTS operation is automatic (assuming the system is operable), it is considered certain to run. (Note: It is assumed that conditions in the reactor building that would result in the fire sprays being triggered would also cause the isolation of SGTS by failing the fusible links on the fire dampers. Thus, accident progressions in which fire sprays are operating are excluded from this case.)

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSGTS:	Impossible	Impossible	Impossible
LSGTS:	Certain	Certain	Certain

CASE #2:

This case covers sequences in which the SGTS has failed (or the fire dampers have closed, rendering the system inoperable) or in which there is no ac power. It is thus certain that SGTS is failed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSGTS:	Certain	Certain	Certain
LSGTS:	Impossible	Impossible	Impossible

QUESTION #104: Does H_2 burn in the reactor building after vessel breach?

This question determines whether hydrogen (and carbon monoxide, which is also released by the corium-concrete interaction) burns in the reactor building following vessel breach. Late combustion of hydrogen released earlier is not considered (if a burn were going to occur, it would have already). Combustion in the reactor building can impact fission product retention by increasing the level of reactor building bypass and actuating the fire protection systems. This question is analogous to Question #62. The computer analyses described thereunder apply directly to this case, especially the MELCOR calculations [15], in which carbon monoxide combustion was considered.

This is a Type 2 question, and there are two branches:

1. LHB: Combustible gases burn in the reactor building following vessel breach.

2. nLHB: There is no burning of combustible gases in the reactor building following vessel breach.

CASE #1:

In this case, the reactor building has been completely bypassed. This could have resulted either from venting the containment through the ILRT line or from previous hydrogen burns. In either case, it is assumed that combustible mixture cannot accumulate if the building has been bypassed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LHB:	Impossible	Impossible	Impossible
nLHB:	Certain	Certain	Certain

CASE #2:

In this case, the core debris has been quenched. If anything, only steam would be released to the reactor building. Combustion of hydrogen, which may have been released earlier, has already been considered. Thus, burns are considered impossible for this case.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LHB:	Impossible	Impossible	Impossible
nLHB:	Certain	Certain	Certain

CASE #3:

In this case, there is leakage (Level 2) of combustible gases from the containment subsequent to vessel breach. The Standby Gas Treatment System is operating however, and depending on the rate at which combustible gases are produced, they may or may not accumulate to flammable concentrations in the reactor building. The SGTS will reduce the chance of combustion by removing some portion of the gas from the reactor building, thus lessening the probability that there will be flammable concentrations. As in Case 4 of Question #62, a 50-50 chance of combustion is assumed, reflecting lack of any basis for making an assessment of the probability.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LHB:	Unlikely	Indeterminate	Likely
nLHB:	Likely	Indeterminate	Unlikely

CASE #4:

In this case, there is significant leakage from the containment, and the Standby Gas Treatment System is not operating. Without SGTS, combustible gases may accumulate to flammable levels, depending on the production rate inside the containment. The IDCOR model predicts that flammable concentrations are not attained in the reactor building. However, the Source Term Code Package models predict combustible concentrations are readily attained. It would seem that combustion is certain based on the STCP predictions which form the basis of the central estimate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LHB:	Indeterminate	Certain	Certain
nLHB:	Indeterminate	Impossible	Impossible

CASE #5:

This case covers sequences in which the containment retains its integrity throughout the accident. No significant quantities of combustible gas are released to the reactor building, so there is no chance of combustion.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LHB:	Impossible	Impossible	Impossible
nLHB:	Certain	Certain	Certain

QUESTION #105: What is the level of late RB bypass?

This question considers the impact of possible combustible gas burns in the reactor building on its capability for fission product retention. Burning within the building will increase the probability of bypass and also the level of bypass. Doorways and hatches could be blown out, and leakage through the building walls could be increased. In addition, burns will accelerate the flow of gas through the building, decreasing the residence time. This question is

340

BEST COPY AVAILABLE

completely analogous to Question #63. More complete documentation of the considerations used in the quantification is given thereunder.

This is a Type 2 question, and there are four branches:

1. nLRBB: The reactor building retains its integrity throughout the accident.

2. LRBB2: There is leakage flow from the reactor building but all the leakage from the primary containment passes through the building (to the refueling bay, then to the environment).

3. LRBB3: The reactor building is essentially bypassed. Either primary containment leakage is directly to the refueling bay or there is sufficient damage to the reactor building that the retention along the release pathway would be similar to that case.

4. LRBB4: The reactor building is completely bypassed. There would be no significant fission product retention.

CASE #1:

The reactor building was bypassed even without the effects of later combustible gas burns. The building is certain to remain bypassed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Impossible	Impossible	Impossible
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Certain	Certain	Certain

CASE #2:

In this case, the reactor building had a Level 3 bypass prior to the burning of combustible gases. This case is quantified exactly as is Case #2 of Question #63.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Impossible	Impossible	Impossible

LRBB3:	Indeterminate	Unlikely	Impossible
LRBB4:	Indeterminate	Likely	Certain

CASE #3:

In this case, a hydrogen burn occurs. The reactor building was previously intact or leaking through the blow-out panels. This case is identical to Case #3 of Question #63, and the probabilities for reactor building bypass are quantified exactly the same.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Very Likely	Likely	Impossible
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Very Unlikely	Unlikely	Certain

CASE #4:

In this case, the reactor building bypass was previously Level 3. No burn occurred, so the bypass level remains the same.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Impossible	Impossible	Impossible
LRBB3:	Certain	Certain	Certain
LRBB4:	Impossible	Impossible	Impossible

CASE #5:

In this case, there was previously leakage from the reactor building. No burn occurred. The reactor building remains only partially bypassed.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Impossible	Impossible	Impossible
LRBB2:	Certain	Certain	Certain
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Impossible	Impossible	Impossible

CASE #6:

In this case, the SGTS is handling the leakage from the primary containment. There is no bypass.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLRBB:	Certain	Certain	Certain
LRBB2:	Impossible	Impossible	Impossible
LRBB3:	Impossible	Impossible	Impossible
LRBB4:	Impossible	Impossible	Impossible

QUESTION #106: Are fire systems operating late?

This question determines the impact of possible hydrogen burns in the reactor building on the operability of the fire sprays. The description of Question #60 above includes a description of the fire spray system. This question is completely analogous to Question #64.

This is a Type 2 question, and there are two branches:

1. LFireS: The reactor building fire sprays are running during at least some portion of the period following vessel breach.

2. nLFireS: The fire sprays are not actuated during the accident.

CASE #1:

In this case, the fire sprays were not actuated prior to vessel breach. Reactor building temperature remained below the threshold at which the fire sprays are triggered after vessel breach, and no combustible gas burns occurred. The fire spray will thus never be actuated.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Impossible	Impossible	Impossible
nLFireS:	Certain	Certain	Certain

CASE #2:

In this case, the fire sprays were not actuated prior to the hydrogen burn that occurs following vessel breach. However, because the drywell failure location is assumed to be in the drywell head closure seal the burn is assumed to occur above the refueling floor (described in the discussion above). A burn in this location will not cause the fire sprays to come on.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Impossible	Impossible	Impossible
nLFireS:	Certain	Certain	Certain

CASE #3:

This case covers sequences in which the fire sprays were operating prior to vessel breach or were activated subsequent to vessel breach either by the hot gases released from the containment or by a subsequent hydrogen burn.

QUANTIFICATION

	Optimistic	Central	Pessimistic
LFireS:	Certain	Certain	Certain
nLFireS:	Impossible	Impossible	Impossible

QUESTION #107: Does standby gas treatment work late?

This question determines whether the Standby Gas Treatment System (SGTS) continues to operate throughout the accident. Hydrogen burns in the containment (except if they occur above the refueling floor) will cause SGTS to isolate since the fusible links in the duct work will fail, resulting in the closure of the fire dampers.

This is a Type 2 question, and there are two branches:

1. nLSGTS: The Standby Gas Treatment System (SGTS) does not operate during the latter portion of the accident following vessel breach.

2. LSGTS: The Standby Gas Treatment System (SGTS) continues to operate throughout the accident.

CASE #1:

In this case, the SGTS was operating after vessel breach, and no hydrogen burns occurred in the reactor building. The fire dampers will thus remain open, and SGTS is certain to continue to operate.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSGTS:	Impossible	Impossible	Impossible
LSGTS:	Certain	Certain	Certain

CASE #2:

This case covers both scenarios in which the SGTS has previously failed and scenarios in which the fire sprays were triggered late in the accident. For either scenario, SGTS is inoperable.

QUANTIFICATION

	Optimistic	Central	Pessimistic
nLSGTS:	Certain	Certain	Certain
LSGTS:	Impossible	Impossible	Impossible

References

1. A. B. Benjamin, et al., *Containment Event Analysis for Postulated Severe Accidents: Surry Power Station, Unit 1*, NUREG/CR-4700, Volume 1, Sandia National Laboratories, February, 1987.

2. V. Behr, et al., *Containment Event Analysis for Postulated Severe Accidents: Sequoyah Power Station, Unit 1*, NUREG/CR-4700, Volume 2, Sandia National Laboratories, February, 1987.

3. A. M. Kolaczkowski, et al., *Analysis of Core Damage Frequency from Internal Events: Peach Bottom, Unit 2*, NUREG/CR-4550, Volume 4, Sandia National Laboratories, October, 1986.

4. R. M. Harrington and S. A. Hodge, *ATWS at Brown's Ferry Unit One - Accident Sequence Analysis*, NUREG/CR-3470, ORNL/TM-8902, June, 1984.

5. H. S. Blackman, et al., *Containment Venting Analysis for the Peach Bottom Nuclear Power Plant (Draft)*, NUREG/CR-4696, Idaho National Engineering Laboratory, July, 1986.

6. C. N. Amos, et al., *Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Peach Bottom, Unit 2*, NUREG/CR-4551, Volume 3, Sandia National Laboratories, May, 1987.

7. L. G. Greimann, et al., *Reliability Analysis of Steel Containment Strength*, NUREG/CR-2442, June, 1982.

8. Reference Deleted

9. R. S. Denning, et al., *Radionuclide Release Calculations for Selected Severe Accident Scenarios: BWR, Mark 1 Design*, NUREG/CR-4624, Volume 1, Battelle's Columbus Division, July, 1986.

10. R. S. Denning, et al., *Radionuclide Release Calculation for Selected Severe Accident Scenarios: BWR, Mark III Design*, NUREG/CR-4624, Volume 4, Battelle's Columbus Division, July, 1986.

11. R. M. Harrington, et al., *Station Blackout at Brown's Ferry Unit One - Accident Sequence Analysis*, NUREG/CR-2181, August, 1982.

12. Personal Communication, S. A. Hodge, Oak Ridge National Laboratory, March, 1986.

13. MAAP Analysis of the La Salle County Station for Postulated Severe Accidents, Fauske & Associates, Inc., FAI/85-41, October, 1985.

14. C. D. Leigh, et al., MELCOR Validation Results, SAND86-2128, Sandia National Laboratories, October, 1986.

15. S. E. Dingman, et al., Analysis of Peach Bottom Station Blackout with MELCOR, SAND86-2129, Sandia National Laboratories, October, 1986.

16. Steam Explosion Review Group. A Review of the Current Understanding of the Potential for Containment Failure Arising from An In-Vessel Steam Explosion, NUREG/CR-1116, U.S. Nuclear Regulatory Commission: February, 1985.

17. IDCOR Task 17.5, Letter Report from Chicago Bridge & Iron Company, November, 1985.

18. J. A. Gieseke, et al., Radionuclide Release Under Specific LWR Accident Conditions, Volume 2, BWR, Mark 1 Design, BMI-2104, Battelle Columbus Laboratories, July, 1984.

APPENDIX C

Supporting Calculations For The SARRP Peach Bottom Event Tree

1. Reactor Vessel and Containment Parameters

Table 1
Reactor Vessel

Parameter	Symbol	Value	Units
Power	pw	3293	MW
Initial Water Inventory	M_{LV}	$6.7\text{x}10^5$	lbm
Initial Steam Inventory	M_{GV}	$2.4\text{x}10^4$	lbm
Operating Pressure	Pv	1050	psia
Liquid Enthalpy	h_{fv}	550	Btu/lbm
Steam Enthalpy	h_{gv}	1190	Btu/lbm
Liquid Density	V_{fv}	46.3	lbm/ft^3
Steam Density	V_{gv}	2.24	lbm/ft^3
Vessel Volume	V_v	$2.5\text{x}10^4$	ft^3
Mass of Zircaloy	Mzr	$1.44\text{x}10^5$	lbm
Mass of UO	Muo	$3.51\text{x}10^5$	lbm

Table 2
Containment

Parameter	Symbol	Value	Units
Drywell Free Volume	V_D	$1.59\text{x}10^5$	ft^3
Wetwell Free Volume	V_W	$1.19\text{x}10^5$	ft^3
Total Volume	V_c	$2.78\text{x}10^5$	ft^3
Mass in Suppression Pool	M_p	$8.7\text{x}10^6$	lbm

SUPPORTING CALCULATIONS

Calculations used in providing information for the CET for the following questions are included:

Question #25, Cases 1 & 2
Question #27, Case 4
Question #28, Case 3
Question #29, Case 2
Question #36, Cases 4 & 5
Question #42, Cases 4, 6, 9, & 10
Question #43, Cases 2, 4, & 5
Question #73, Case 2
Question #74, Cases 2 & 3
Question #75, All Cases

QUESTION #25: Does an SRV tailpipe vacuum breaker stick open?

ASEP quantification of failure probability: 0.03 per demand. A factor of 10 reduction is assumed to account for design modifications. This reduction is considered optimistic. The unreduced value is pessimistic. No frequency reduction for multiple demands. To quantify probability of stuck-open tailpipe vacuum breaker (SOTVB), the number of demands is estimated for each case.

CASE #1: Fast Station Blackout, no depressurization.

- From Reference 1, Figure 4.24, estimate 4.7×10^5 lbm (M_B) boil-off to core uncovery.
- From Reference 1, Table 4.2 (TB1), additional 7.1×10^4 lbm boil-off to core sump.
- Assume boil-off from core collapse forces several (~ 3) SRVs open for an extended period (add six challenges to account for this).
- Calculate steam relief from 1 SRV opening.
  - Assume steam volume in RPV is 1.5×10^4 ft^3 on average during the boil-off ($=V_s$).
  - Assume 3×10^5 lbm saturated water in vessel on the average ($=M_V$).
  - Assume SRV lift drops RPV pressure from 1150 psia to 1100 psia.

BEST COPY AVAILABLE

- Steam generated by flashing:

$$M_f = \frac{M_V \left[h_f\ (1150) - h_f\ (1100)\right]}{h_{fg}\ (1125)}$$

$$M_f = 3 \times 10^5 \text{ lbm} \times \frac{565 \text{ Btu/lbm} - 557 \text{ Btu/lbm}}{636 \text{ Btu/lbm}}$$

$$M_f = 3.8 \times 10^3 \text{ lbm}$$

- Steam released due to steam dome expansion: (flashing neglected).
  $Me = Vs\ (\rho 1150 - \rho 1100)$
  $Me = 1.5 \times 10^4\ ft^3\ (45.6\ lbm/ft^3 - 45.17\ lbm/ft^3)$
  $Me = 6.45 \times 10^3\ lbm$

- Total steam release: Mt
  $Mt = Me + Ms$
  $Mt = 3.8 \times 10^3\ lbm + 6.45 \times 10^3\ lbm$
  $Mt = 10.3 \times 10^3\ lbm/demand$

- Total mass release during boil-off = 4.75×10^5 lbm + 7.1×10^4 lbm = 5.4×10^5 lbm

- Number of demands:
  $N = 5.4 \times 10^5\ lbm/(10.3 \times 10^3\ lbm/demand)$
  N = 52 demands

- Failure probability:
  52 demand x 3×10^{-3}/demand = 0.16

In the pessimistic estimate the probability of failure is taken as 0.03 per demand (the ASEP value). The resultant failure probability is unity (1.0). The central estimate is taken as the mean of the optimistic and pessimistic values:

$$P_C = \frac{0.16 + 1.0}{2} = 0.6$$

This evaluation introduces a bias toward the optimistic quantification, this is judged to be reasonable.

CASE #2: Blackout (Initial HPCI and/or RCIC) or TW.

These sequences involve high pressure boil-off similar to Case #1 above. In addition, the lowest set SRV will cycle from the time the MSIVs close until the time the operators

BEST COPY AVAILABLE

330

take control of vessel pressure. This is assumed to occur when the suppression pool temperature reaches 120°F. This temperature selection is based on the procedures which instruct the operators to control vessel pressure within the HTCL (described under Question #17).

Calculation of SRV cycles prior to pool temperature reaching 120°F:

- Assume 90°F initial temperature (30°F rise)
- Heat added:

$$H = c_p M_p \times \Delta T = (1\ \text{Btu/lbm.°F})\ (8.7\text{x}10^6\ \text{lbm})\ (30\text{°F})$$
$$H = 2.61 \times 10^8\ \text{Btu}$$

- Average enthalpy of steam released from vessel
  h_g = 1106.2 Btu/lbm (saturated @ 1125 psia)
- Average suppression pool enthalpy:
  h_p = 73 Btu/lbm (saturated @ 105°F)
- Calculate steam mass released from vessel to raise pool temperature:

$$M = \frac{H}{h_g - h_p} = \frac{2.61 \times 10^8\ \text{Btu}}{(11062.2 - 73)\ \text{Btu/lbm}}$$

$$M = 2.5 \times 10^5\ \text{lbm}$$

- Calculate number of SRV pops

$$N = \frac{M}{M_t} = \frac{2.5 \times 10^5\ \text{lbm}}{10.3 \times 10^3\ \text{lbm/d}} = 25\ \text{demands}$$

(M_t from Case #1 above)

25 demands + 52 demands (from Case #1)
N = 77 demands

- Failure probability:
  77 demands x $3\text{x}10^{-3}$/demand = 0.23

- Central Estimate:

$$P_C = \frac{1 + 0.23}{2} = 0.6$$

QUESTION #27: Does the containment fail before core degradation?

This question is answered yes or no except for the case in which RPV depressurization causes loss of low pressure injection, HPCI and RCIC are failed and CRD flow cannot maintain adequate injection. This occurs in the TC-core vulnerable sequence (Case #4). In this situation, the pressurization of the containment may or may not cause failure prior to core damage.

The following calculation of containment pressurization during RPV boil-off applies to Case #4. The calculation applies to the central estimate only.

- As in Case #1 of Question #25 assume 4.7×10^5 lbm of water boiled-off from RPV (M_B).

- Assume injection lost with containment pressure at 110 psia (T_P = 335°F).

- Iterative calculation is used to estimate containment pressure:

  - Assume M_e lbm of pool evaporates
  - Assume P' is the new containment pressure

  Then:

$$h_g M_B = C_p M_p \left[T_{sat}(P') - T_p\right] + M_e h_g (P')$$

$$P' = \frac{(M_e/M)\, R\, [T_{sat}(P') + 460°R]}{V_C} + P_o$$

M = 18 lbm/lb mole (molecular weight of water)

R = 1544 ft-lbf/lb-mole°R

P_o = 110 psia

Substituting known values:

$$(1106\ \text{Btu/lbm})\,(4.7\times10^5\text{lbm}) = (1\ \text{Btu/lbm°F})\,(8.7\times10^6\text{lbm}) \times [T_{sat}(P') - 335°F] + M_e h_g (P')$$

352

BEST COPY AVAILABLE

$$P' - P_O = \frac{(M_e/18)\ 1545\ [T_{sat}(P') + 460°R]}{2.78 \times 10^5\ ft^3}$$

Combining:

$$5.2x10^8 = 8.7x10^6\ [Tsat(P') - 335] + \frac{3.24 \times 10^3 (P' - 110)(144)}{T_{sat}(P') + 460}\ h_g\ (P')$$

P' (psia)	$T_{sat}(P')$ (F)	$h_g(P')$ Btu/lbm	r.h.s. Btu
130	347	1193	$1.2x10^8$
140	353	1194	$1.8x10^8$
190	378	1199	$4.2x10^8$
220	390	1201	$5.5x10^8$

Interpolating:

$$P' = 190 + \frac{5.2 \times 10^8 - 4.2 \times 10^8}{5.5 \times 10^8 - 4.2 \times 10^8}\ (30)$$

$$P' = 213 \text{ psia}$$

Since the central estimate of containment failure pressure is 132 psia, the probability of the containment surviving is negligible.

QUESTION #28: What is the location of early containment failure?

Only Case #3 considers location (drywell versus wetwell) of containment failure given that an overpressure failure occurs. The central estimate of the probability of wetwell failure is taken as the mean value of those given by the SARRP expert reviewers [3]. The weighting factors given to wetwell failure for issue 7B were:

Reviewer	Weighting Factors
1	0
2	0.2
3	0.3
4	0.1
5	0.2
6	0.3
7	0.1
	1.2

The mean weighting factor (probability) is: 1.2/7 = 0.17
The central estimate is 0.2.

QUESTION #29: For drywell failures, is the failure at the head seal?

Only Case #2 considers the location (head versus knuckle) given drywell failure.

Using responses to Issue 7B [3]:

Reviewer	Probability of Drywell Failure (DF)	Weighting of Head Failure (HF)	Probability of Head Failure Given Drywell Failure (HF/DF)
1	1	0.6	0.6
2	0.8	0.4	0.5
3	0.7	0.25	0.36
4	0.9	0.2	0.22
5	0.8	0.25	0.31
6	0.7	0.3	0.43
7	0.9	0.5	0.56
			2.98

The mean weighting factor (probability) is: P = 2.98/7 = 0.43 Central estimate of head failure probability given drywell failure is thus 0.4.

QUESTION #36: What is the containment pressure prior to core degradation?

CASE #1: Described in text.

CASE #2: Described in text.

CASE #3: Described in text.

CASE #4: Assume large break (without drywell spray). Add steam partial pressure resulting from 2 hours of decay heat added to the suppression pool. (This uses MARCH assumption of wetwell - pool equilibrium.)

- Integrated decay heat at 2 hrs = 113.8 full-power seconds =

  F_D

  $$\Delta T = \frac{P_W}{C_p}\frac{F_D}{M_p}$$

  $$\Delta T = \frac{(3293 \times 10^6\ W)\ (113.8\ s)\ (9.49 \times 10^{-4}\ \frac{Btu}{W\text{-}s})}{(1\ Btu/lbm\ °F)\ (8.7 \times 10^6\ lbm)}$$

- Assume initial temperature 121°F ([2], Table 6.4) the initial steam partial pressure would be 1.7 psia.
- Final temperature, 162°F (121 plus 41) the partial pressure would be 5.0 psia.
- Pressure rise is 5 - 1.7 or 3.3 psia.
- Add to 35 psia for Case #3. The result is 38 psia. This is rounded up to 40 psia. (This gives significant allowance for concrete degassing.)

CASE #5: Station Blackout, no injection.

- Mass boiled to pool prior to core uncovery:
  $M_B = 4.7\text{x}10^5$ lbm (Question #25, Case #1 above)

  Resulting pool temperature rise:

  $$\Delta Tp = \frac{h_g}{C_p}\frac{M_B}{M_p} = \frac{(1106\ Btu/lbm)\ (4.7 \times 10^5\ lbm)}{(1\ Btu/lbm \bullet °F)\ (8.7 \times 10^6\ lbm)}$$

  $$\Delta Tp = 60°\ F$$

- From Reference 1, Table 4.3, temperature rise in suppression pool from core uncovery to start melt is 6°F.

- Assume initial pool temperature (at start of melt) is 156°F. The corresponding steam partial pressure is 4.3 psia.

- Calculate partial pressure of noncondensables assuming the drywell temperature rises from 135°F (operating temperature) to 350°F (approximate result from loss of drywell coolers). Assume wetwell temperature is 156°F.

Initial moles of non-condensable:

$$N = \frac{1}{R}\left[\frac{P_{ID}V_D}{T_{ID}} + \frac{P_{IW}V_W}{T_{IW}}\right]^*$$

$$N = 6.47 \times 10\text{-}4\ \frac{\text{lb-mole}^\circ R}{\text{ft-lbf}}\left[\frac{(2117\ \text{lbf/ft}^2)(1.59 \times 10^5\ \text{ft}^3)}{595^\circ R} + \frac{(2117\ \text{lbf/ft}^2)(1.19 \times 10^5\ \text{ft}^3)}{550^\circ R}\right]$$

$$N = 662\ \text{lb-moles}$$

At the elevated temperatures:

$$N = \frac{P}{R}\left[\frac{V_D}{T_D} + \frac{V_W}{T_W}\right]$$

$$P = \frac{NR}{\frac{V_D}{T_D} + \frac{V_W}{T_W}}$$

$$P = \frac{(662)(1545)}{\frac{1.59 \times 10^5}{810} + \frac{1.19 \times 10^5}{616}} = 2626\ \frac{\text{lbf}}{\text{ft}^2}$$

$$P = 18.2\ \text{psi}$$

*Subscripts: ID = initial drywell, D = drywell,
IW = initial wetwell, W = wetwell.

- Total pressure:

  18.2 + 4.3 psi = 22.5 psia
  Use 23 psia as central estimate

CASES #6 - #12: Described in text.

QUESTION #42: What is the hydrogen generation and pressure rise during core degradation?

CASES #1 through #3: Described in text.

CASE #4: LOCA (Large or Medium)

- From Reference 2, Tables 6.3 and 6.4 (AE):

- At containment failure the pressure has risen 100 psi (from start of melt) and 34% of the Zircaloy has reacted.

- The total quantity of Zr reacted is 36% (to bottom head failure).

- For the pessimistic estimate (see text).

$$\Delta P = \Delta P \text{ (34\% Zr) } \times \frac{.36}{.34} = 106 \text{ psi}$$

(Assumes pressure rise is proportional to Zr oxidation.)

- For the central estimate (neglect adiabatic compression: 36% of Zr reacted:

  0.36 (Mzr)/(91.2 lbm/lb-mole Zr) = 570 lb-mole Zr

  570 lb mole Zr x (2 lb-mole H_2/lb-mole Zr)

  = 1140 lb-mole H_2

- Assume suppression pool temperature is 130°F ([1], Table 6.4, AE). Assumed equal to wetwell temperature when core melt is gradual.

- Assume drywell temperature 2000°F ([1], Table 6.4, AE). Calculate pressure:

  - Following Case #5 of Question #36

BEST COPY AVAILABLE

$$P = \frac{NR}{\frac{V_D}{T_D} + \frac{V_W}{T_W}}$$

N = 662 + 1140 = 1802 lb-moles

$$P = \frac{(1802)(1545)}{\frac{1.59 \times 10^5}{2460} + \frac{1.19 \times 10^5}{590}}$$

P = 72.6 psi

Initial pressure was 30 psi. Thus the pressure rise would be 43 psi. Round to 45 psi.

- For the optimistic estimate assume 5% clad reacted, thus 158 lb-mole of H_2 produced (typical of MAAP code calculations, [4])

  N = 662 + 158 = 820 lb-mole

  Using the above equation,

  P = 33.1 psi

  The base pressure (Question #36, Case #1) is 17 psi thus the pressure rise is 16.

CASE #6: Deferred to after CASE #9 (see below).

CASE #9: Transient, vessel at high pressure.

- Central estimate of pressure rise is 24 psi ([1], see text)
- Calculate hydrogen distribution for STCP calculation:
  - Zr oxidation is 25% (1583 lbm of H_2 produced 792 lb-moles)
  - Calculate H_2 partial pressure in containment: ([1] Table 4.3 - TB1) pressure at bottom head dry-out = 51.8 psia, pool temperature = 244°F. Thus steam partial pressure is 26.8 psi.

    Partial pressure H_2:

51.8 - 14.7 - 26.8 = 10.3 psi

Calculate quantity of H_2 in containment:

Drywell temperature = 196°F

Wetwell temperature = 244°F

$$N = \frac{P}{R}\left[\frac{V_D}{T_D} + \frac{V_W}{T_W}\right]$$

$$N = \frac{(10.3\ \mathrm{lbf/in}^2)(144\ \mathrm{ft}^2/\mathrm{in}^2)}{1545\ \mathrm{ft\text{-}lbf/lb\text{-}mole°R}}$$

$$\frac{1.59 \times 10^5 \mathrm{ft}^3}{656°\mathrm{R}} + \frac{1.19 \times 10^5 \mathrm{ft}^5}{704°\mathrm{R}}$$

N = 395 lb-moles

Thus 50% remains in the vessel.

Volume fraction in vessel: (assume 2500°F average)

$$N_V = \frac{P_V\ V_V}{RT_V} = \frac{(1050\ \mathrm{lbf/in}^2)(144\ \mathrm{in}^2/\mathrm{ft}^2)(2.5 \times 104\ \mathrm{ft}^3)}{(1545\ \mathrm{ft\text{-}lbf/lb\text{-}mole\ R})(2960°\mathrm{R})}$$

N_V = 827 lb-moles

N = 397 lb-moles

Volume fraction = 48%

- Optimistic estimate:

  Assume 5.2% Zr reacted (MAAP calculation of TQUV at LaSalle [4])

  - Total H production:

    792 (0.052/0.25) = 165 lb-moles

    Quantity in-vessel:

    48% (0.052/0.25) = 10% or 82.7 lb-moles

    Quantity in containment: 82 lb-moles

BEST COPY AVAILABLE

$$P_H = \frac{NR}{\frac{V_D}{T_D} + \frac{V_W}{T_W}}$$

Assume temperatures are same as for central estimate:

$$P_H + \frac{(83 \text{ lb-mole})(1545 \text{ ft-lbf/lb-mole°R})}{\frac{1.59 \times 10^5 \text{ft}^3}{656\text{°R}} + \frac{1.19 \times 10^5 \text{ft}^3}{704\text{°R}}}$$

$$P = 309 \text{ lbf/ft}^2 = 2 \text{ psi}$$

- Pool temperature rise 14°F for STCP calculation (TB1)

$$14 \text{ R } \frac{0.052}{0.25} = 3\text{°F}$$

Pool temperature = 233°F

Psat = psia

- Relative to central estimate:

  Hydrogen partial pressure 8 psi lower

  Steam partial pressure 4.8 psi lower

  Optimistic estimate is 13 psi lower than central or 11 psi.

- Pessimistic Estimate

  Based on MARCON results [5] 34.4 psia in containment at start of core degradation 106.2 psia in containment immediately prior to vessel failure. Thus, pressure rise is 72 psi.

CASE #6: (Deferred from above) transient with open SRV tailpipe vacuum breaker

- Assume drywell temperature is 550°F rather than STCP calculated value of 196°F.

  - From Case #5 of Question #36 there are 622 lb-moles of noncondensable gas initially in containment.

- From Case #6 (this question) the central estimate is 395 lb-moles of H in containment. Thus assume 1060 lb-moles in containment.

- Calculate noncondensable pressure.

$$P = \frac{NR}{\frac{V_D}{T_D} + \frac{V_W}{T_W}}$$

$$P = \frac{(1060 \text{ lb-mole})(1545 \text{ ft-lbf.lb-mole°R})}{\frac{1.59 \times 10^5 \text{ft}^3}{1010°R} + \frac{1.19 \times 10^5 \text{ft}^3}{704°R}}$$

$$P = 5020 \text{ lbf/ft}^2 = 34.8 \text{ psi}$$

- Steam partial pressure is 26.8 psi (assuming same as Case #9).

  - Total pressure = 61.6 was 51.8 for STCP so pressure rise is 10 psi higher than for the case without a stuck open SRV tailpipe vacuum breaker.

    - Use 34 psi as the central estimate of the pressure rise.

- For the optimistic evaluation repeat the above calculation assuming:

  T_D = 400°F (reasonable lower bound)

  N_H = 82 lb-moles (H_2 in containment from Case #9 optimistic evaluation).

  - Pressure due to noncondensables

    P = 22.6 psi (744 lb-moles of noncondensable gas in containment)

  - Partial pressure of steam = 22 psi (233°F pool temperature as in Case #9).

  - Total pressure = 44.66 psi, 17 psi lower than calculated for the central estimate.

  - Use 17 psi as the optimistic evaluation of the pressure rise (34-17).

- For the pessimistic evaluation calculate the pressure corresponding to the MARCON calculation ([5] and Case #9b of this question) assume a 700°F drywell temperature (reasonable upper bound).

MARCON calculated parameter:

Drywell temperature, T_D = 306°F

Wetwell temperature, T_W = 335°F

Total pressure = 106.2 psia

- Calculate moles of gas in containment

$$N = \frac{P}{R}\left[\frac{V_D}{T_D} + \frac{V_W}{T_W}\right]$$

$$N = \frac{(106.2\ \text{lbf/in}^2)(144\ \text{in}^2/\text{ft}^2)}{1545\ \text{ft-lbf/lb-mole}^\circ\text{R}}$$

$$\frac{1.59 \times 10^5\,\text{ft}^3}{766^\circ\text{R}} + \frac{1.19 \times 10^5\,\text{ft}^3}{795^\circ\text{R}}$$

N = 3536 lb-moles

- Calculate pressure assuming 1160°R drywell temperature.

$$P = \frac{NR}{\frac{V_D}{T_D} + \frac{V_W}{T_W}}$$

P = 132 psi

- Without stuck-open tailpipe vacuum breaker pressure was 106 psi. Thus, 26 psi (132-106) is added to Case #6 pessimistic result. Pressure rise of 98 psi is taken as pessimistic evaluation (72+26).

362

CASE #10: Transient with vessel depressurized.

The calculation of pressure rise is similar to Case #9, above. The following changes are made to account for depressurization of the vessel:

1. Pool temperature raised 4°F. This is the temperature rise due to vessel blowdown calculated for the TB1 sequence using the STCP ([1], Table 4.3).
2. Vessel hydrogen inventory added to containment.
3. Vessel volume added to containment (drywell) volume.

- Central Estimate
  - 662 moles noncondensables initially in containment
  - 792 moles of hydrogen added
  - 28.8 psi partial pressure of steam (248°F in wetwell) = P_s
  - Assume drywell temperature 196°F as in Case #6
  - Assume vessel temperature 2500°F
  - Calculate pressure of noncondensables

$$P_C = \frac{NR}{\frac{V_D}{T_D} + \frac{V_W}{T_W} + \frac{V_V}{T_V}}$$

$$P_C = \frac{(1450 \text{ lb-moles})(1545 \text{ ft-lbf/lb-mole°R})}{\frac{1.59 \times 10^5 \text{ft}^3}{656°\text{R}} + \frac{1.19 \times 10^5 \text{ft}^3}{704°\text{R}} + \frac{2.5 \times 10^4}{2960°\text{R}}}$$

Pc = 37 psi

- Add steam partial pressure

  $P = P_C + P_s = 37 + 29 = 66$ psi

BEST COPY AVAILABLE

In Case #9 the central estimate of containment pressure was 51.8 (based on STCP calculation).

66 - 51.8 = 14 psi higher than Case #9 pressure rise.

Case #9 value was 24 psi; use 38 psi for this case.

- Optimistic Estimate

Following the central evaluation (same temperatures) but assume 165 moles of hydrogen (see Case #9) for a total of 827 lb-moles of noncondensables.

$$P_c = 21 \text{ psi}$$

This is 16 psi less than central evaluation. Thus, pressure rise is 22 psi (38-16).

- Pessimistic Estimate

MARCON calculates 817 lb-moles of hydrogen in containment [5]. Assume ratio of in-vessel to exvessel is same as STCP calculation.

$$\frac{792 \text{ (total for STCP)}}{397 \text{ (ex-vessel for STCP}} \times 817 \text{ (ex-vessel for MARCON)} = 1626 \text{ moles } H_2 \text{ Total}$$

Thus 809 moles of H_2 were in the vessel, 3536 moles (all gases) in the containment.

Calculate pressure resulting from adding hydrogen, increasing pool temperature to 339°F (4°F higher than MARCON).

$$P = \frac{[(809 + 3536)\text{lb-moles}](1545 \frac{\text{ft-lbf}}{\text{lb-mole°R}})}{\frac{1.59 \times 10^5 \text{ft}^3}{766\text{°R}} + \frac{1.19 \times 10^5 \text{ft}^3}{799\text{°R}} + \frac{2.4 \times 10^4 \text{ft}^3}{2960\text{°R}}}$$

$$P = 128 \text{ psi}$$

Pressurized case pressure was 106 psi (Case #9). Thus increase was 22 psi due to depressurization. Case #9 pressure rise was 72 psi thus 98 psi is used for this case (72 + 22).

QUESTION #43: Does at least one drywell vacuum breaker stick open?

The failure probability per demand recommended by ASEP is that used in WASH-1400 of 1.1×10^{-4} per demand. This is a median value; the mean value is 3.3×10^{-4} per demand. The failure probability is reduced a factor of 10 for subsequent demands. The number of challenges is estimated as described in the text.

CASE #1: Described in text.

CASE #2: 110 psi containment pressure; 1 demand per 1 psi increment.

$$P_F = 3.3 \times 10^{-4} + 3.3 \times 10^{-5} (109) = 0.004$$

(central estimate)

CASE #3: Described in text.

CASE #4:

- Optimistic - Assume 20 demands

$$P_F = 3.3 \times 10^{-4} + 3.3 \times 10^{-5} (19)$$

$$P_F = 1 \times 10^{-3}$$

- Central - Assume 46 demands

$$P_F = 3.3 \times 10^{-4} + 3.3 \times 10^{-5} (45)$$

$$P_F = 2 \times 10^{-3}$$

- Pessimistic - Assume 113 demands

$$P_F = 3.3 \times 10^{-3} + 3.3 \times 10^{-4} (112)$$

$$P_F = 0.04$$

CASE #5: High vessel pressure. Assume no reduction in failure probability for subsequent demands (vacuum breaker will open more violently).

- Optimistic Estimate - Assume 20 demands

$$P_F = 20 (3.3 \times 10^{-4}) = 7 \times 10^{-3}$$

- Central Estimate - Assume 40 demands

$$P_F = 40 (3.3 \times 10^{-4}) = 1.4 \times 10^{-2}$$

- Pessimistic Estimate - Assume 97 demands

  $P_F = 97\ (3.3 \times 10^{-3}) = 0.3$

CASE #6: Described in text.

CASE #7: Described in text.

QUESTION #73: What is the pressure rise due to [RPV] blowdown?

CASE #1: Described in text.

CASE #2: Vessel blowdown from high pressure, no direct heating.

- Estimate pressure rise with hand calculation:
  - Initial N_2 + air in containment = 662 lb-moles

    (Question #36, Case #5)
  - Hydrogen added during core degradation = 395 lb-moles

    (Question #42, Case #9)

    Initial drywell temperature: 196°F (STCP, [1])

  Initial pool (wetwell) temperature 244°F (STCP, [1])

  Vessel contains: 397 lb-moles H_2 (48%)
  430 lb-moles H_2 (52%)

    (Question #42, Case #9)

- Assume average temperature of gas in-vessel is 2737°F (STCP, [1], Table 4.2)

Worst Case:

- Gas in vessel expands isothermally into the containment

$$P_V = \frac{NRT}{V} = \frac{(927\ \text{lb-moles})(1545\ \text{ft-lbf/lb-mole°R})(3197°\text{R})}{(3.03 \times 10^5 \text{ft}^3)(144\ \text{in}^2/\text{ft}^2)}$$

P_V = 93.6 psi = partial pressure exerted by gas from the vessel

More Realistic Case:

- Assume steam passing to pool is condensed and that hydrogen is cooled to the pool temperature.

Estimate quantity passing to pool:

Volume that gas released occupies:

$$V_G = \frac{NRT}{P} = \frac{(827\ \text{lb-moles})(1545\ \text{ft-lbf/lb-mole R})(3197°\text{R})}{P\ (144\ \text{in}^2/\text{ft}^2)}$$

$V_G = 2.84 \times 10^7/P$

Volume of gas passing to pool = $V_G - V_D$

Quantity of steam not condensed:

$= 430\ \text{lb-moles}\ [1- ((V_G-V_D)/V_G)]$
$= 430\ \text{lb-moles}\ (V_D/V_G)$
$= 430\ \text{lb-moles}\ (6.49 \times 10^{-3}\ P)$
$= 2.79\ P$

$$P_{NC} = \frac{NR}{\frac{V_D}{T_D} + \frac{V_W}{T_W}}$$

(V_D = drywell + vessel)
(T_W assumed 250°F)

$$P_{NC} = \frac{(662+395+397+2.79\ P)\ \text{lb-moles}\ (1545\ \text{ft-lbf/lb-mole°R})}{\left[\frac{1.84 \times 10^5 \text{ft}^3}{3197°\text{R}} + \frac{1.19 \times 10^5}{710°\text{R}}\right]\ (144\ \text{in}^2/\text{ft}^2)}$$

$$P_{NC} = \frac{(145 + 2.79\ P)(1545)}{225\ (144)}$$

$P = P_{NC}$ + partial pressure of steam

$P = P_{NC}$ + 30 psi

$$P_{NC} = \frac{(1538 + 2.79\ P_{NC})(1545)}{3.24 \times 10^4}$$

$P_{NC} = 73.3 + 0.133\ P_{NC}$

$0.87\ P_{NC} = 73.3$

$P_{NC} = 85$ psi

Total pressure = 85 psi + 30 psi = 115 psia

This value is close to the STCP calculation (129 psi [1]). However, the drywell temperature is 336°F not 2737°F. If 336°F were assumed above, the pressure calculated would be 78 psi which would be equivalent to a 22 psi pressure rise. Thus the central estimate of a 38 psi pressure rise is intermediate to these simple estimates.

CASE #3: Described in text.

QUESTION #74: What is the pressure rise due to debris quenching?

CASE #1: Described in text.

CASE #2: Use the responses of the SARRP expert reviewers. ([3] Issue 11, Case #2)

Reviewer:	1	2	3	4	5	6	7
Pressure Rise			Weighting Factor				
15 psi	0	0	0	0.9	0.1	0	0
30 psi	0.9	1	1	0.1	0.9	1	0.9
45 psi	0.1	0	0	0	0	0	0.1
60 psi	0	0	0	0	0	0	0
75 psi	0	0	0	0	0	0	0
>120 psi	0	0	0	0	0	0	0

Cumulative weight for each level:

Pressure Rise	Weight
15 psi	1.9
30 psi	5.0
45 psi	0.2
60 psi	0
75 psi	0
>120 psi	0

Thus 30 psi pressure rise is used as the central estimate.

SARRP Calculation:

Assume drywell contains maximum water:

30 ft diameter floor, covered 1.5 ft deep.

368

Mass = $\rho \pi R^2 d$

= $(62.4 \text{ lbm/ft}^3)\ \pi\ (15 \text{ ft})^2\ \ (1.5 \text{ ft})$

= 6.62×10^4 lbm

STCP calculation of core quenching at slump ([1], Table 4.2, TB1).

At core collapse:

Mass of water in vessel = $M_V = 1.81 \times 10^5$ lbm

Average core temperature = T_C = 3407°F

Fraction Zr reacted = $F_Z = 0.22$

At bottom head dryout:

$Mv = 4.17 \times 10^4$ lbm

T_C = 2428°F

$F_Z = 0.25$

Core collapse evaporated:

$\Delta M_V = 1.81 \times 10^5 - 4.17 \times 10^4 = 1.39 \times 10^5$ lbm

ΔT_C = 3407°F - 2248°F = 979°F

$\Delta F_Z = 0.25 - 0.22 = 0.03$

- Assume bottom head failure evaporates all the water on the drywell floor. Ratio Zr oxidation based on mass of water evaporated. Assume 6.62×10^4 lbm of saturated steam is produced (3678 lb-moles).

- Hydrogen production

$$0.03\ \frac{6.62 \times 10^4 \text{lbm}}{1.39 \times 10^5 \text{lbm}} = 0.014 \text{ of Zr}$$

$$0.014\ \left[(1.44 \times 10^5 \text{ lbm Zr})\left(91.2\ \frac{\text{lbm Zr}}{\text{lb-mole Zr}}\right)\right]$$

$$2\ \frac{\text{lb mole } H_2}{\text{lb mole Zr}} = 45 \text{ lb-mole } H_2$$

Calculate pressure rise (same approach as Case #2 of Question #73).

Estimate quantity of steam which passes to the suppression pool.

- Assume drywell temperature 330°F (saturated at 100 psi)
- Assume wetwell temperature 250°F (roughly STCP value).

$$V_G = \frac{NRT}{P} = \frac{(3678 \text{ lb-mole})(1545 \text{ ft-lbf/lb-mole R})(790°R)}{P\ (144 \text{ in}^2/\text{ft}^2)}$$

$V_G = 3.12 \times 10^7/P$

Volume of steam going to pool = $V_G - V_D$

Quantity of steam not condensed:

$= 3678 \text{ lb-mole } [1 - (V_G - V_D/V_G)]$

$= 3678 \text{ lb-mole } (V_D/V_G)$

$= 3678 \text{ lb-mole } (5.1 \times 10^{-3}\ P)$

$= 18.8\ P$

$$P_{NC} = \frac{NR}{\dfrac{V_D}{T_D} + \dfrac{V_W}{T_W}}$$

$$P_{NC} = \frac{(1454 + 18.8\ P)\text{lb-mole}(1545 \text{ ft-lbf/lb-mole°R})}{\left[\dfrac{1.59 \times 10^3}{790°R} + \dfrac{1.19 \times 10^5}{710°R}\right] \quad 144\ \dfrac{\text{ft}^2}{\text{in}^2}}$$

$$P_{NC} = \frac{(2017 + 18.8\ P_{NC})(1545)}{5.31 \times 10^4}$$

$$P_{NC} = 129 \text{ psi}$$

Total pressure = 129 psi + 30 psi = 159 psia

This is the instantaneous pressure, assuming no condensation in the drywell is much larger than what is actually expected. The 45 lb-moles of hydrogen assumed to be produced would result in negligible added pressurization. The pressure rise from this calculation would be 69 psi, enough to almost certainly fail the containment. It is not used since it is judged a severe overestimate.

CASE #3: Pressure rise due to the boil-off of 500 gallons of water in the drywell sumps. The expert responses were ([3], Issue 11a, no bypass).

Reviewer:	1	2	3	4	5	6	7
Pressure Rise			Weighting Factor				
5 psi	0.2	0.5	0.5	0.2	0.6	0.3	0.6
15 psi	0.5	0.5	0.5	0.6	0.4	0.7	0.4
30 psi	0.3	0	0	0.2	0	0	0
45 psi	0	0	0	0	0	0	0

Cumulative weight for each level:

Pressure Rise	Weight
5 psi	2.9
15 psi	3.6
30 psi	0.5
45 psi	0

Thus, the central estimate is 15 psi.

Using the simple SARRP model (as in Case #2):

500 gal x 0.1337 ft^3/gal x 62.4 lbm/ft^3/(18 lbm/lb-mole) = 231 lb-mole, P_{NC} = 82 psi

Total pressure = 82 psi + 30 psi = 112 psi

The pressure rise would be 22 psi. Again this procedure overestimates the pressure rise since the amount of steam assumed to condense is the absolute lower bound.

QUESTION #75: Is the suppression pool bypassed or (drywell) vents not cleared at vessel breach (and what is the effective pressure rise)?

CASE #1: Ex-vessel steam explosion, suppression pool "saturated" (as defined in Question #58) but not bypassed.

Since ex-vessel steam explosions are very unlikely, the pressure rise in the containment which results from a steam explosion has been only crudely estimated. These estimates are based on hand calculations which extrapolate results of calculations made by an early version of the MARCH computer code. These calculations are not repeated here since they do not impact calculated risk. Essentially these calculations estimate the pressure rise resulting from 100% of the water within the confines of the vessel pedestal (25 foot in diameter by 2 foot deep) being evaporated into the drywell instantaneously.

CASE #2: Ex-vessel steam explosion, suppression pool not bypassed

Same comment applies as for Case #1.

CASE #3: Direct heating, suppression pool "saturated" but not bypassed.

Based on a simple calculation of direct heating loads for Peach Bottom, performed for SARRP by R. Gasser of SNL, any significant amount of direct heating (>10% of the core debris participating) would result in pressures sufficiently high to assure containment failure. Thus, regardless of whether or not the vents clear, the pressure rise of direct heating is always set sufficiently large to assure containment failure.

CASE #4: Direct heating, suppression pool not bypassed - see discussion under Case #3.

CASE #5: Direct heating and ex-vessel steam explosion suppression pool "saturated" but not bypassed - see discussion under Case #3.

CASE #6: Direct heating and ex-vessel steam explosion suppression pool not bypassed - see discussion under Case #3.

CASE #7: Water on drywell floor suppression pool "saturated" but not bypassed.

From Case #2 of Question #73 the pressure rise is 30 psi with the pool subcooled. The pressure rise for this case is the added pressure from the pool being "saturated" plus the added pressure from vessel blowdown to a saturated pool.

- o Calculate energy deposited in the pool
  - Vessel blowdown (assume 100% of static enthalpy goes to pool). Assuming average temperature within the vessel is 2400°F:

    400 lb-moles H_2, $h_{H2} = 2 \times 10^4$ Btu/lb-mole

    400 lb-moles H_2O, $h_g = 3.5 \times 10^4$ Btu/lb-mole

    Total energy deposited pool:

(400 lb-mole)[(2 x 10^4 Btu/lb-mole) + 3.5 x 10^4 Btu/lb-mole] = 2.2 x 10^7 Btu

  - Evaporation of water on the drywell floor (see Case #2 of Question #37 above).

    45 lb-mole of H_2

    6.62 x 10^4 lbm of steam produced

    Assume 2400°F is also the average temperature at which steam and hydrogen are produced (debris quenched from 2700°F).

(45 lb-mole)(2 x 10^4 Btu/lb-mole)
+ (6.62 x 10^4 lbm)(3.5 x 10^4 Btu/lb-mole)/18 lbm/lb-mole = 1.3 x 10^8 Btu

- o Calculate pool enthalpy (specific) increase
  (1.3 x 10^8 Btu + 2.2 x 10^7 Btu)/8.6 x 10^6 lbm = 18 Btu/lbm

  @180°F d Psat/dh = 0.23 psia/(Btu/lbm)(subcooled)

  @250°F d Psat/dh = 0.53 psia/(Btu/lbm)(saturated)

  Thus saturated versus subcooled gives 0.3 psi-lbm/Btu difference in steam partial pressure.

(18 Btu/lbm) 0.3 psi-lbm/Btu = 6 psi
use -15% for optimistic (3 psi)
use +100% for pessimistic (12 psi)

CASE #8: Suppression pool subcooled and not bypassed.

No additional pressure rise.

CASE #9: Suppression pool bypassed but containment failed.

No pressure rise.

CASE #10: Drywell floor wet, RPV at high pressure, suppression pool bypassed.

This is containment loading Issue 11 Cases 4 and 5 (pressure rises added) in Reference 3. The mean of the expert responses would indicate that the total pressure rise for this case should be 105 psi. Subtracting the 38 psi resulting from vessel blowdown (Question #73, Case #2) and 30 psi resulting from debris quenching (Question #74, Case #2) the pressure rise for this case would be 37 psi. However, the calculation below indicated that this would be optimistic.

SARRP Calculation:

From Question #73, Case #2, estimate 3678 lb-moles of steam would be released from debris quenching and 45 lb-moles of H_2. As in Question #42, Case #9, 827 lb-moles of gas are released from the vessel. The total gas release to containment is thus:

3678 + 45 + 827 = 4550 lb-moles

The resulting pressure (1057 lb-moles of gas present initially: N_2 + air + H_2) is given by: (assume 500°F final temperature)

$$P = \frac{NRT}{V_c + V_v}$$

$$P = \frac{(4550 + 1057 \text{ lb-mole})(1545 \text{ ft-lbf/lb-mole}°\text{R})(960°\text{R})}{(3.03 \times 10^5 \text{ft}^3)(144 \text{ in}^2/\text{ft}^2}$$

P = 191 psia

Subtracting the 68 psi previously accounted for the pressure rise for this case is 120 psi. The optimistic and pessimistic values are ±30% from the base case.

CASE #11: Same as Case #10 but with the RPV depressurized.

The quantity of gas in the containment would be identical to the previous case but the temperature would be lower since the steam and hydrogen from the vessel would be released gradually and most (or all) would pass through the suppression pool.

Repeating the above calculation assuming a 300°F mean temperature in containment, the added pressure rise is 90 psi.

CASE #12: Suppression pool bypassed, RPV depressurized, drywell floor dry.

This case is just the added pressure rise due to boil-off of water in the drywell sumps. From Case #2 of Issue 11a [3] the expert weighting factors were:

Reviewer: Pressure Rise	1	2	3	4	5	6	7
5 psi	0	0.5	0.4	0	0.6	0	0.4
15 psi	0.3	0.5	0.4	0.5	0.4	0.4	0.6
30 psi	0.5	0	0.2	0.5	0	0.6	0
45 psi	0.2	0	0	0	0	0	0

Cumulative weight for each level:

Pressure Rise	Weight
5 psi	1.9
15 psi	3.1
30 psi	1.8
45 psi	0.2

Thus, the central estimate is 15 psi. No correction for previous 10 psi pressure rise (Question #73 Case #3) was made since this would result in an overly optimistic estimate.

CASE #13: Suppression pool bypassed, RPV pressurized, drywell floor dry.

Use the Expert responses--Issue 11, Case #4 [3].

BEST COPY AVAILABLE

Reviewer: Pressure Rise	1	2	3	4	5	6	7
30 psi	0	0	-	0	0	-	-
45 psi	0	0	-	0	0	-	-
60 psi	0.1	0.05	-	0.09	0.15	-	-
75 psi	0.7	0.9	-	0.9	0.8	-	-
>120 psi	0.2	0.05	-	0.01	0.05	-	-

Cumulative Weight:

Pressure Rise	Weight
30 psi	0
45 psi	0
60 psi	0.39
75 psi	3.3
>120 psi	0.31

Thus 75 psi is the consensus level. Subtracting the 38 psi previously accounted for (Question #73) and adding 15 psi for the water in the drywell sumps (previous case). We employ 45 psi as the central estimate (52 psi rounded downward to the nearest level is deemed appropriate).

REFERENCES

1. R. S. Denning et al., "Radionuclide Release Calculations For Selected Severe Accident Scenarios," Volume 1, BWR, Mark I Design, Battelle Columbus Division, November 18, 1985.

2. J. A. Gieseke et al., "Radionuclide Release Under Specific LWR Accident Conditions," Volume 2, BWR, Mark I Design, BMI-2104, Battelle Columbus Laboratories, July 1984.

3. C. N. Amos et al., Evaluation of Severe Accident Risks and The Potential For Risk Reduction: Peach Bottom, Unit 2, U. S. Nuclear Regulatory Commission Report, NUREG/CR-4551, Volume 3, Sandia National Laboratories, Albuquerque, NM, May 1987.

4. MAAP Analysis of The LaSalle County Station For Postulated Severe Accidents, Fauske & Associates, Inc., FAI/85-41, October 1985.

5. Personal Communication, S. A. Hodge, Oak Ridge National Laboratory, March 1986.

6. R. M. Harrington and S. A. Hodge, "ATWS at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-3740, ORNL/TM-8902, June 1984.

7. L. D. Greimann et al., "Reliability Analysis of Steel Containment Strength," NUREG/CR-2442, June 1982.

8. R. M. Harrington et al., "Station Blackout at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-2181, August 1982.

APPENDIX D

Sample Event Tree Problem

A sample event tree problem has been constructed to facilitate the discussion of how input for the event tree program is structured. Before discussing this sample problem, some of the capabilities of the program are discussed.

The event tree program, EVNTRE, was written to do the bookkeeping of very large event trees that were necessary to properly address problems in the Severe Accident Risk Reduction Program (SARRP). The event trees developed in SARRP typically involved 50 to 100 "top questions" or "nodes." This in itself established a need for an automated bookkeeping procedure for the trees. In addition, without making very conservative assumptions nearly all of the branch point probabilities were dependent upon the occurrence or absence of prior events in the tree--something very few other past event tree analyses have allowed for. Not only were most of the branch point probabilities dependent upon the path taken in the event tree up to that point, some of the probabilities needed to be calculated internally. That is, the user did not know a priori what the probability would be. This was most frequently the case for the probability of containment failure. In these instances, pressure loads were compared to estimates of a containment failure pressure distribution to obtain an estimate of containment failure probability. Keeping these points in mind, several key features of the program will be discussed.

As mentioned above, most of the questions involve user supplied branch point probabilities. However some questions utilize the capability of the program to internally calculate the branch point probability. The way this is performed is certain "parameters" which have had values specified at previous nodes are added together and compared to "reference parameters" which have a value specified at the current node. Take for example the probability of containment failure due to an early hydrogen burn. This is dependent upon the baseline pressure in containment prior to the burn, the probability of a hydrogen burn occurring, the pressure rise due to the burn, and containment failure pressure. Typically the value of baseline pressure and pressure rise due to the burn would be supplied at prior nodes as "parameters." The probability of the burn occurring would be input as a branch point probability at an earlier node (most likely the same one that supplied the value of pressure rise due to the burn). Then at the node which addresses containment failure due to early hydrogen burn, the program would be instructed to add the values of baseline pressure and hydrogen-burn pressure rise and compare the sum to a distribution of containment failure pressure to obtain a branch point probability. The containment failure pressure (a "reference parameter") is specified through input for that node.

From the discussion above, it becomes apparent that there are at least three different types of questions for the event tree. They are those which the user supplies only branch point probabilities, those for which the user supplies "parameter" values as well as branch point probabilities, and those which the user supplies a list of "parameters" to be added and "reference parameters" to which the sum is to be compared. In fact, there are six different types of questions distinguished by the program. The remaining three types arise by considering the three types mentioned above as being either independent of or dependent on prior events in the tree. A summary of the different question types is shown below in Table D.1.

Table D.1

Summary of the Different Question Types for the Event Tree Program

	Dependency Upon Prior Events	
	Independent	Dependent
User Supplies Branch Point Probabilities Only	Type 1	Type 2
User Supplies Branch Point Probabilities and Parameter Values	Type 3	Type 4
User Specifies a List of Parameters to be Summed and Compared to Reference Parameters to Obtain Branch Point Probabilities	Type 5	Type 6

For those questions which are dependent upon prior events, different "dependency cases" are defined for which different sets of branch point probabilities are supplied. The "dependency cases" are described by specifying branches which must have been or must not have been taken for a series of key questions. To specify a branch that must have been taken, the question number and the branch number are supplied to the program in the set of data for that "dependency case." To specify that a branch must not have been taken, the negative of the branch number is supplied to indicate a "not" condition. Using this strategy, many different "dependency cases" can be specified.

Marching through the event tree and finding the probability of each of the pathways would be useless without some means of condensing all of the different outcomes into some more tangible form. Thus the program was written to bin all of the different outcomes, in some user specified fashion, for the output. This binning is performed based upon certain key characteristics by which the user would like to be able to differentiate the output. For example, for the SARRP analysis we were interested in the fraction of the sequence initiating frequency which would result in different modes of early and late containment failure, with containment sprays operating or not, with different levels of secondary containment bypass, etc.

Based on these needs the program was set up to have characteristically different "dimensions" for binning. These "dimensions" are things such as those mentioned in the previous paragraph. Within each of these "dimensions," different bins are specified. For example, in the early containment failure category there may be several different levels of containment leakage as well as gross failures due to several different causes. In the case of containment sprays, commonly considered bins were different combinations of early and late spray failure and success. The binning information for the SARRP analyses was chosen to adequately describe scenarios such that fission product source terms could be assigned to the scenarios without requiring an overly conservative binning of the outcomes.

The binning is completely user-specified. Input is prepared to specify the number and title of each of the different "dimensions" and the number and title of each of the bins within each of the dimensions. Then information is provided identifying combinations of key questions and branches which must or must not have been taken for an outcome to fall within each of the respective bins in each "dimension." The output then consists of aggregate probabilities of occurrence of different combinations of the various "dimensional bins."

Table D.2 provides a sample input deck constructed solely for the purpose of familiarizing the novice with the EVNTRE input. It deals with early failure of containment in a subatmospheric PWR containment due to an early hydrogen burn. The discussion of the input follows in Table D.3 with reference to line numbers in the sample input found in Table D.2.

The event tree processor EVNTRE has been extended to allow the use of boolean expressions in the specification of pathway dependencies for particular cases in the tree quantification and in the binning of pathways through the tree. Boolean expressions facilitate the building and readability of large trees and have been used extensively in the event tree for Peach Bottom accident progression analysis. Table D-4 provides an example of a boolean expression that combines cases two and four in Question #7 of the sample event tree in Table D-2.

Table D.2

Sample Problem Input

Line No.

Input

```
 1 SAMPLE - Early failure due to H2 burning in S3D at a PWR
 2       3     1.0     1.0     1.0       8  1.0E-04 1.0E-07
 3     Nwal  Pinit   Pinit   Pinit  Nquest  Prntol   Toler
 4   1 Is there pre-existing leakage or isolation failure?
 5       4     EE3     EE2     EE1     EE0
 6       1       1       2       3       4
 7           0.000   0.000   0.013   0.987
 8           0.000   0.002   0.028   0.970
 9           0.000   0.002   0.051   0.947
10   2 Initial break size
11       4       A      S1      S2      S3
12       1       1       2       3       4
13           0.000   0.000   0.000   1.000
14           0.000   0.000   0.000   1.000
15           0.000   0.000   0.000   1.000
16   3 Induced failure in primary system?
17       5       A      S1      S2      S3    none
18       2       1       2       3       4       5
19       2
20       1       2
21               4
22              S3
23           0.000   0.000   1.000   0.000   0.000
24           0.250   0.000   0.250   0.000   0.500
25           0.000   0.000   0.000   0.000   1.000
26         Otherwise
27           0.000   0.000   0.000   0.000   1.000
28           0.000   0.000   0.000   0.000   1.000
29           0.000   0.000   0.000   0.000   1.000
30   4 Sprays on early?
31       2     ESp   noESp
32       2       1       2
33       5
34       1       1
35               1
36             EE3
37           0.000   1.000
38           0.000   1.000
39           0.000   1.000
40       2       1       2
41               2       1
42             EE2       A
43           1.000   0.000
44           1.000   0.000
45           1.000   0.000
```

Table D.2 (Continued)

Sample Problem Input

Line No.	Input			
46	2	1	2	
47		2	2	
48		EE2	S1	
49		0.900	0.100	
50		0.900	0.100	
51		0.900	0.100	
52	3	1	2	2
53		2	-1	-2
54		EE2	noA	noS1
55		0.000	1.000	
56		0.000	1.000	
57		0.000	1.000	
58		Otherwise		
59		1.000	0.000	
60		1.000	0.000	
61		1.000	0.000	
62	5 Early baseline pressure			
63	1	Pbase		
64	4	1		
65	5			
66	3	1	1	4
67		-1	-2	1
68		noEE3	noEE2	ESp
69		1.000		
70		1.000		
71		1.000		
72	1			
73	1	20.000		
74		20.000		
75		20.000	EPbase	
76	3	1	1	4
77		-1	-2	2
78		noEE3	noEE2	noESp
79		1.000		
80		1.000		
81		1.000		
82	1			
83	1	40.000		
84		40.000		
85		40.000	EPbase	

Table D.2 (Continued)

Sample Problem Input

<u>Line No.</u>

<u>Input</u>

```
   86       2       1       4
   87               2       1
   88             EE2     ESp
   89           1.000
   90           1.000
   91           1.000
   92       1
   93       1    15.0
   94            15.0
   95            15.0  EPbase
   96       2       1       4
   97               2       2
   98             EE2   noESp
   99           1.000
  100           1.000
  101           1.000
  102       1
  103       1    20.0
  104            20.0
  105            20.0  EPbase
  106         Otherwise
  107           1.000
  108           1.000
  109           1.000
  110       1
  111       1    14.7
  112            14.7
  113            14.7  EPbase
  114   6 Maximum break size in primary system
  115       4       A      S1      S2      S3
  116       2       1       2       3       4
  117       7
  118       2       2       3
  119               4       5
  120              S3    none
  121           0.000   0.000   0.000   1.000
  122           0.000   0.000   0.000   1.000
  123           0.000   0.000   0.000   1.000
  124       2       2       3
  125               3       5
  126              S2    none
  127           0.000   0.000   1.000   0.000
  128           0.000   0.000   1.000   0.000
  129           0.000   0.000   1.000   0.000
```

Table D.2 (Continued)

Sample Problem Input

```
Line No.

                              Input

 130       2       2       3
 131               2       5
 132              S1    none
 133           0.000   1.000   0.000   0.000
 134           0.000   1.000   0.000   0.000
 135           0.000   1.000   0.000   0.000
 136       1       3
 137               4
 138              S3
 139           0.000   0.000   0.000   1.000
 140           0.000   0.000   0.000   1.000
 141           0.000   0.000   0.000   1.000
 142       1       3
 143               3
 144              S2
 145           0.000   0.000   1.000   0.000
 146           0.000   0.000   1.000   0.000
 147           0.000   0.000   1.000   0.000
 148       1       3
 149               2
 150              S1
 151           0.000   1.000   0.000   0.000
 152           0.000   1.000   0.000   0.000
 153           0.000   1.000   0.000   0.000
 154          Otherwise
 155           1.000   0.000   0.000   0.000
 156           1.000   0.000   0.000   0.000
 157           1.000   0.000   0.000   0.000
 158     7 Hydrogen burn before vessel breach?
 159       2     EHB   noEHB
 160       4       1       2
 161       6
 162       4       1       1       4       6
 163              -1      -2       1       1
 164           noEE3   noEE2     ESp       A
 165           0.000   1.000
 166           0.100   0.900
 167           0.100   0.900
 168       1
 169       2     0.0     0.0
 170            25.0     0.0
 171            50.0     0.0   DP-EH
 172       4       1       1       4       6
 173              -1      -2       1       2
```

Table D.2 (Continued)

Sample Problem Input

Line No.	Input				
174		noEE3	noEE2	ESp	S1	
175		0.000	1.000			
176		0.000	1.000			
177		0.100	0.900			
178	1					
179	2	0.0	0.0			
180		0.0	0.0			
181		35.0	0.0	DP-EH		
182	5	1	1	4	6	6
183		-1	-2	1	-1	-2
184		noEE3	noEE2	ESp	noA	noS1
185		0.000	1.000			
186		0.000	1.000			
187		0.000	1.000			
188	1					
189	2	0.0	0.0			
190		0.0	0.0			
191		0.0	0.0	DP-EH		
192	3	1	4	6		
193		2	1	1		
194		EE2	ESp	A		
195		0.000	1.000			
196		0.000	1.000			
197		0.100	0.900			
198	1					
199	2	0.0	0.0			
200		0.0	0.0			
201		35.0	0.0	DP-EH		
202	3	1	4	6		
203		2	1	2		
204		EE2	ESp	S1		
205		0.000	1.000			
206		0.000	1.000			
207		0.100	0.900			
208	1					
209	2	0.0	0.0			
210		0.0	0.0			
211		25.0	0.0	DP-EH		
212		Otherwise				
213		0.000	1.000			
214		0.000	1.000			
215		0.000	1.000			
216	1					
217	2	0.000	0.000			
218		0.000	0.000			
219		0.000	0.000	DP-EH		

Table D.2 (Concluded)

Sample Problem Input

Line No.

Input

```
220   8 Does containment fail due to an early hydrogen burn?
221      2     EH   noEH
222      5      1      2
223      2      1      2
224        EPbase  DP-EH
225         'AND'
226      'NORMAL'      2    134    2.5
227                         134     15
228                          60     15
229                  CFP   MEAN  SIGMA
230 'SAMPLE Early failure due to H2 burning in S3D at a PWR
231 'OPTIMISTIC'  'CENTRAL'     'PESSIMISTIC'
232   2       'CF' 'Sprays'
233   5  5  'EEO'  'EE1'   'EE2'  'EE3'    'EH'
234   2  1      1      8
235             4      2
236           EEO   noEH
237   2  2      1      8
238             3      2
239           EE1   noEH
240   2  3      1      8
241             2      2
242           EE2   noEH
243   2  4      1      8
244             1      2
245           EE3   noEH
246   1  5             8
247                    1
248                   EH
249   2  2  'ESp' 'noESp'
250   1  1      4
251             1
252           ESp
253   1  2      4
254             2
255         noESp
```

Table D.3

Detailed Discussion of Sample Input

Line in Table D.2	Input Description
1	Dummy title for this set of input. Not actually used by program but made available for the user for quick identification of the input
2-3	Number of quantifications (or walkthroughs) of the tree, the initiating frequency of each of the quantifications (when set equal to one, conditional probabilities are obtained), the number of questions in the tree, the print tolerance (output with aggregate probability less than the print tolerance is not found in the printed output), and the cutoff tolerance in the tree (if the probability of a pathway is reduced below the cutoff tolerance, calculations cease for that portion of the tree). The titles in line three are not used by the program but facilitate the user in preparing the input.
4	Question #1 is concerned with the level of preexisting leakage or isolation failure. The title is not used by the program.
5	This first node in the tree has four branches. They are titled EE3, EE2, EE1, and EE0. EE3 stands for leakage of sufficient size to be equivalent to a gross failure of containment. EE2 is leakage smaller than gross failure (in terms of amounts of fission products released) but still large enough to prevent containment failure due to long term and gradual overpressurization. EE1 is leakage too small to prevent long term overpressurization but larger than design leakage of containment. Design leakage or less is designated by EE0. These titles are not used by the program but are provided for ease of preparing and reading the input.
6	The first number is all that is actually used by the program and denotes this question as being of Type 1 (see earlier discussion of question types). The other numbers are simply the branch numbers for this question which are supplied for ease of reading the input.
7-9	These three lines provide the branch point probabilities for this question for the three different quantifications. Each line supplies the probabilities for a different quantification. In this example (this will be true for all the questions in

Table D.3

Detailed Discussion of Sample Input

Line in Table D.2	Input Description
7-9 (cont.)	this sample input), the first line represents input for an optimistic quantification, the third line for a pessimistic quantification, and the second line represents a central viewpoint.
10	Question #2 addresses the size of the initial break in the primary system.
11-12	The question has four branches (A, S1, S2, S3) and is of Type 1. The branches are "large" (A), "intermediate" (S1), "small" (S2), and "very small" (S3).
13-15	The branch point probabilities for the three different quantifications are given in these three lines.
16	Question #3 is concerned with the possibility of induced failures in the primary system in high pressure scenarios due to natural convection of hot gases throughout the system.
17-18	Question #3 has five branches (four similar to Question #2 and one additional to account for no induced failure) and is of Type 2. Type 2 means the branch point probabilities are dependent upon prior events such that "dependency cases" must be specified.
19	There will be two "dependency cases" specified
20	For this first "dependency case," there is one question-branch pair specified which must either have or have not been taken previously in the tree for the following branch point probabilities to be used for this question. The question of dependency is Question #2.
21	The branch specification is 4. This means (since the branch specification is positive) that Branch 4 must have been taken previously at Question #2 for the following set of branch point probabilities to be used.

Table D.3

Detailed Discussion of Sample Input

Line in Table D.2	Input Description
22	The title of the branch which must have been taken is S3. This title is not used by the program but is supplied for ease of preparation and examination of the input.
23-25	The branch point probabilities for this dependency case for this question for the three different quantifications.
26	This is the second and last dependency case. In fact, it is a catchall case. That is, any paths which did not fit into any of the previous cases use this set of branch point probabilities. (Hence the "otherwise" descriptor.) This line is not actually used by the program but is another clarification feature of the input structure.
27-29	The three sets of branch point probabilities for the "otherwise" catchall case.
30-32	Question #4 addresses early spray operation, it has two branches, and is of Type 2.
33	Five dependency cases are specified for this question.
34-36	The first dependency case has one question-branch pair specified; this is Branch 1 of Question #1 (the EE3 branch).
37-39	The branch point probabilities for spray operation given preexisting leakage or isolation failure equivalent to gross rupture of containment.
40-42	The second dependency case has two question-branch pairs specified; these are Questions #1 and #2 and Branches 2 and 1 respectively, corresponding to preexisting leakage or isolation of Level 2 and a large initial break in the primary system. Both question-branch pairs must have been taken previously for the set of branch point probabilities that follow to be used.
43-45	The branch point probabilities for spray operation for the three different quantifications and the second dependency case.

Table D.3

Detailed Discussion of Sample Input

Line in Table D.2	Input Description
46-51	The third dependency case specification (leakage level 2 and intermediate initial break in the primary system) and the corresponding branch point probabilities.
52-54	The fourth dependency case specifies a leakage level of 2 and an initial break in the primary system other than large or intermediate (signified by the negative branch numbers). Thus either a small or a very small initial break with a leakage level of 2 would fall into this case.
55-57	The branch point probabilities for the fourth dependency case.
58-61	The catchall branch point probabilities for Question #4.
62	Question #5 addresses the early baseline pressure in containment.
63-65	There is only one branch for this question since it is only in the tree to supply the value of the early baseline pressure "parameter." The question is of Type 4, meaning it is dependent upon prior events and that a parameter value is specified. There are five dependency cases specified.
66-71	The first dependency case is specified as Branch 1 of Question #4 and anything but Branch 1 or 2 on Question #1. Since this is a single branch node, the probabilities for all three quantifications are all unity.
72	The value of 1 parameter will be specified in this case. (The value of more than one parameter can be specified as a Type 3 or 4 question.)
73-75	The parameter identification number is 1 and the value of the parameter for the three different quantifications is 20.0.
76-113	Specification of the remaining four dependency cases, and the associated branch point probabilities, parameter identification numbers and parameter values.

Table D.3

Detailed Discussion of Sample Input

Line in Table D.2	Input Description
114	Question #6 combines the information from Questions #2 and #3 to simplify the specification of dependency cases in later questions. No new information is gained by this node, but past information is summed up into a more convenient form.
115-117	Question #6 has four branches and is of Type 2. There are seven dependency cases specified.
118-157	Specification of the seven different dependency cases and the branch point probabilities. Note that since this question serves in a summary manner, the branch point probabilities are all unity or zero and are invariant with the three different quantifications.
158	Question #7 addresses the probability of a hydrogen burn before vessel breach and the magnitude of such a hydrogen burn.
159-161	Question #7 consists of two branches and is of Type 4. Six dependency cases are specified.
162-167	The first dependency case has four question-branch pairs specified. They are "not" Branch 1 or 2 of Question #1 and Branch 1 of both Questions #4 and #6. The three sets of branch point probabilities are also supplied.
168	The value of 1 parameter will be specified for this dependency case.
169-171	The parameter identification number is 2. (This parameter is the pressure rise associated with the hydrogen burn.) The values of the parameter are 0., 25., and 50., for the three different quantifications if Branch 1 is taken. Otherwise the values are all zero, since the other branch involves no combustion. (The value of the parameter for a given branch appears under that branch in order of the different quantifications.)
172-219	Specification of the remaining five dependency cases along with the parameter identification and values.

Table D.3

Detailed Discussion of Sample Input

Line in Table D.2	Input Description
220	Question #8 addresses the probability of containment failure due to an early hydrogen burn.
221-222	Question #8 has two branches (a necessity for this type of question) and is of Type 5. This means it is of the type which compares a sum of specified parameters to some reference parameters and the branch point probability will be internally calculated by the program.
223-224	There are two parameters specified to manipulate. These are Parameters 1 and 2. These are the values of early baseline pressure and early hydrogen burn pressure rise respectively. The titles in line 224 are provided entirely for clarification reasons and are not actually used in the program.
225	The parameters are to be "anded" together. That is they are to be summed. Other options here are 'MAX' and 'MIN' which specify the program to take the maximum or minimum of the specified parameter values respectively.
226-229	The outcome of the above mentioned manipulation of parameter values is to be compared to a normally distributed reference parameter, hence the 'NORMAL' descriptor in the input. (In this case the reference parameter is the containment failure pressure.) There are two reference parameters, each of which has a value specified for the three different quantifications. For this case, these are the mean and the standard distribution of the normal distribution. Another possibility for this type of question is a threshold comparison in which case 'THRESH' would appear instead of 'NORMAL' and only one reference parameter would need to be specified, namely the value of the threshold. The titles supplied in line 229 are for clarification purposes and are not used by the program.
230	Problem title. (Will be printed in the output.)
231	Titles for the three different quantifications. (Will be printed in the output.)

Table D.3

Detailed Discussion of Sample Input

Line in Table D.2	Input Description
232	The number of binning "dimensions" is specified to be two. The titles of these "dimensions" are CF and Sprays.
233	The first dimension consists of five bins. Five "pathways" for binning into these five bins are specified in the input. (The number of pathways must always equal or exceed the number of bins unless a vacuous bin has been defined.) The bin titles for this dimension are EE0, EE1, EE2, EE3, and EH.
234-236	These lines describe the first "pathway." There are two question-branch pairs specified for this pathway and any outcomes of the tree matching this "pathway" criterion are to be placed in bin Number 1. The question branch pairs are Branch 4 of Question #1 and Branch 2 of Question #8.
237-248	Specification of the remaining four pathways. The first three of which utilize two question-branch pairs and the last only one pair.
249	Dimension two consists of two bins and has two "pathways" specified. The titles are ESp and noESp.
250-252	There is one question-branch pair specified for the first "pathway." Any outcomes of the tree which match this outcome will be placed in bin Number 1 of this dimension.
253-255	The second and last "pathway" specification. Outcomes matching this criterion will be placed in bin Number 2 of this dimension.

Table D.4

Example of a Boolean Expression for a Pathway

```
6        4         1         1         6         1         6
         1      *((3    +    4)   *    2    +    2    *    1)
        ESp       EE1       EEO       S1        EE2        A
```

Notes:
1) * => AND; + => OR
2) AND is evaluated before OR
3) The AND operator is optional (i.e., the absence of an operator symbol is interpreted as an AND operator)
4) -1 and /1 are both interpreted as not Branch 1

The boolean expression requires that Question #4 take Branch 1 (ESp) and either Question #6 take Branch 1 (A) and Question #1 take Branch 2 (EE2), or Question #6 take Branch 2 (S1) and Question #1 take Branch 3 (EE1) or Branch 4 (EEO). This boolean expression would allow the combination in Question #7 of Cases 2 (line 172) and 4 (line 192) in the sample event tree given in Table D.2.

APPENDIX E
Listing of EVNTRE Input For Peach Bottom

```
PEACH BOTTOM EVENT TREE           SARRP
    107
     NQ
      3    1.00    1.00    1.00
        TBUX-optTBUX-cenTBUX-pess
  1 What is the initiating event?
      7       A      S1    S2/3   TLOSP    TPCS      TC    IORV
      1       1       2       3       4       5       6       7
          0.000   0.000   0.000   1.000   0.000   0.000   0.000
          0.000   0.000   0.000   1.000   0.000   0.000   0.000
          0.000   0.000   0.000   1.000   0.000   0.000   0.000
  2 What is the initial break location?
      5     RPV   Rcirc     MSL      FW     SRV
      1       1       2       3       4       5
          0.000   0.000   0.000   0.000   1.000
          0.000   0.000   0.000   0.000   1.000
          0.000   0.000   0.000   0.000   1.000
  3 What is the level of pre-existing leakage or isolation failure?
      5   E1L2D   E1L1D   E1L2W   E1L1W    E1L0
      1       1       2       3       4       5
          0.000   0.000   0.000   0.000   1.000
          0.000   0.000   0.000   0.000   1.000
          0.000   0.000   0.000   0.000   1.000
  4 Is there a loss of offsite power?
      2  E1-LOP  E1nLOP
      1       1       2         ALL TLOSP'S => 1,0; OTHERS => 0,1
          1.000   0.000
          1.000   0.000
          1.000   0.000
  5 Is there a Station Blackout (Diesel generators fail)?
      2      SB     nSB
      1       1       2         ALL TB'S => 1,0; OTHERS => 0,1
          1.000   0.000
          1.000   0.000
          1.000   0.000
  6 Is dc power not available?
      2   E1fDC   E1-DC         TBUX & DCUX => 1,0; OTHERS => 0,1
      1       1       2
          1.000   0.000
          1.000   0.000
          1.000   0.000
  7 For TC, does SLC fail to inject?
      3  E1fSLC  E1-SLC      na   TCS~ => 1,0,0; TC/S~ => 0,1,0;
      1       1       2       3   OTHERS => 0,0,1
          0.000   0.000   1.000
          0.000   0.000   1.000
          0.000   0.000   1.000
  8 Does a S/RV stick open early?
      2 E1-SORV E1nSORV         TBP & TBUP => 1,0; OTHERS => 0,1
      1       1       2
          0.000   1.000
          0.000   1.000
          0.000   1.000
  9 Do the HPCI and RCIC systems fail to inject?
      3  E1fHPC  E1rHPC  E1-HPC          ALL U'S => 1,0,0;
      1       1       2       3          OTHERS => 0,0,1
          1.000   0.000   0.000
          1.000   0.000   0.000
          1.000   0.000   0.000
 10 Does the CRD hydraulic system inject?
      3  E1fCRD  E1rCRD  E1-CRD          TB => 0,1,0; OTHERS => 0,0,1
      1       1       2       3
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
 11 Do the LPCS and LPCI systems fail?
      4  E1fLPC  E1rLPC  E1aLPC  E1-LPC   TB'S => 0,1,0,0; ALL V'S NOT T
      1       1       2       3       4   => 1,0,0,0; OTHERS => 0,0,1,0
          0.000   1.000   0.000   0.000
          0.000   1.000   0.000   0.000
          0.000   1.000   0.000   0.000
 12 Do the RHR systems fail?
      3  E1fRHR  E1rRHR  E1-RHR   TB'S => 0,1,0; ALL W'S
      1       1       2       3   => 1,0,0; OTHERS => 0,0,1
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
 13 Does the condensate system fail?
      3 E1fCOND E1rCOND E1aCOND   TB'S => 0,1,0; ALL V'S NOT TC & V23'S
      1       1       2       3   => 1,0,0; OTHERS => 0,0,1
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
 14 Does HPSW fail in a mode that would preclude injection?
      3 E1fHPSW E1rHPSW E1aHPSW   TB'S => 0,1,0; ALL V'S NOT TC & V23'S
      1       1       2       3   => 1,0,0; OTHERS => 0,0,1
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
 15 Is there a failure that precludes drywell cooler operation?
      3  E1fDWC  E1rDWC  E1aDWC   TB'S => 0,1,0
      1       1       2       3
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
 16 Is ADS blocked or not called upon?
      2  E1nADS  E1-ADS          TC*/A & *X => 1,0; OTHERS => 0,1
      1       1       2
          1.000   0.000
          1.000   0.000
          1.000   0.000
 17 Does the RPV remain pressurized?
      2  E2nDeP  E2-DeP
      2       1       2
      9
```

```
     3       1       1      16
             1   +   2   +   2
             A      S1  E1-ADS
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1       8
             1
        E1SORV                  Depressurization helps re-seat valve
         0.000   1.000
         0.001   0.999
         0.100   0.900
     2       5       9
             1       3
            SB  E1-HPC          HTCL procedures
         0.000   1.000
         0.001   0.999
         0.100   0.900
     1       6
             1
         E1fDC                  ASEP assumes all 5 batteries fail
         1.000   0.000
         1.000   0.000
         1.000   0.000
     5       5       1       1       9      11
             1   +(( 4 +       5)      -3      -4)
            SB   TLOSP    TPCS nE1-HPC  nE1LPC
         0.000   1.000
         0.010   0.990
         0.500   0.500
     5       1       1       9      11      10
            (4 +       5)       3      -4      -3
         TLOSP    TPCS  E1-HPC  nE1LPC  nE1CRD
         0.000   1.000
         0.010   0.990
         0.500   0.500
     6       1       1       9      11      12      10
            (4 +       5)       3      -4       1       3
         TLOSP    TPCS  E1-HPC  nE1LPC  E1fRHR  E1-CRD
         0.001   0.999
         0.010   0.990
         0.500   0.500
     1       1          TCSX & TCUX => 1,0 ; OTHER TC'S => 0,1
             6
            TC
         1.000   0.000
         1.000   0.000
         1.000   0.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
18 For TC, is inadequate level maintained with LPI?
     3      TL TAF/OSC      na
     2       1       2       3
     5
     2       1      16
             6       2
            TC  E1-ADS           ** TCSA'S
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     2       1       7
             6       2
            TC  E1-SLC           ** TC/S'S (e.g. TCUX, TCUV)
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     2       1      17
             6       2
            TC  E2-DeP           ** TCSV => 1,0,0; OTHER TC => 0,1,0
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     1       1
             6
            TC                   ** TCSX **
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
       Otherwise
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
19 What type of sequence is this (summary of plant damage)?
    10      AE      AW     S1E     S1W TQUV/FTB      TB      TW
       TC-fDeP   TC-TL   TC-CV
     2     1\8     2\9    3\10       4       5       6       7
    11
     2       1      11
             1       1
             A  E1fLPC
         1.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   0.000
         1.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   0.000
         1.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   0.000
     2       1      12
             1      -3
             A nE1-RHR
         0.000   1.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   0.000
         0.000   1.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   0.000
         0.000   1.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   0.000
     3       1      11      14
```

```
              2         1         1
             S1    E1fLPC   E1fHPSW
          0.000     0.000     1.000     0.000     0.000     0.000     0.000
          0.000     0.000     0.000
          0.000     0.000     1.000     0.000     0.000     0.000     0.000
          0.000     0.000     0.000
          0.000     0.000     1.000     0.000     0.000     0.000     0.000
          0.000     0.000     0.000
    2         1        12
              2        -3
             S1   nE1-RHR
          0.000     0.000     0.000     1.000     0.000     0.000     0.000
          0.000     0.000     0.000
          0.000     0.000     0.000     1.000     0.000     0.000     0.000
          0.000     0.000     0.000
          0.000     0.000     0.000     1.000     0.000     0.000     0.000
          0.000     0.000     0.000
    7         1         1         1         9        11        11        10
             -3        -6        -7         1        -3        -4        -3
          nS2/3       nTC     nIORV    E1fHPC   nE1aLPC   nE1-LPC   nE1-CRD
          0.000     0.000     0.000     0.000     1.000     0.000     0.000
          0.000     0.000     0.000
          0.000     0.000     0.000     0.000     1.000     0.000     0.000
          0.000     0.000     0.000
          0.000     0.000     0.000     0.000     1.000     0.000     0.000
          0.000     0.000     0.000
    1         5
              1
             SB
          0.000     0.000     0.000     0.000     0.000     1.000     0.000
          0.000     0.000     0.000
          0.000     0.000     0.000     0.000     0.000     1.000     0.000
          0.000     0.000     0.000
          0.000     0.000     0.000     0.000     0.000     1.000     0.000
          0.000     0.000     0.000
    3         1         1        12
             -3        -6        -3
          nS2/3       nTC   nE1-RHR
          0.000     0.000     0.000     0.000     0.000     0.000     1.000
          0.000     0.000     0.000
          0.000     0.000     0.000     0.000     0.000     0.000     1.000
          0.000     0.000     0.000
          0.000     0.000     0.000     0.000     0.000     0.000     1.000
          0.000     0.000     0.000
    2         1        17
              6         1
             TC    E2nDeP
          0.000     0.000     0.000     0.000     0.000     0.000     0.000
          1.000     0.000     0.000
          0.000     0.000     0.000     0.000     0.000     0.000     0.000
          1.000     0.000     0.000
          0.000     0.000     0.000     0.000     0.000     0.000     0.000
          1.000     0.000     0.000
    2         1        18
```

```
             6       1
            TC    E2TL
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   1.000   0.000
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   1.000   0.000
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   1.000   0.000
     2       1      18
             6       2
            TC E2TAF/OC
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   1.000
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   1.000
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   1.000
       Otherwise       *** SHOULD NEVER BE USED ***
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   1.000
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   1.000
         0.000   0.000   0.000   0.000   0.000   0.000   0.000
         0.000   0.000   1.000
20 Is there an LP system break induced by power cycling (ATWS)?
     2   PCyBk  nPCyBk
     2       1       2
     4
     2       1      16
             6       2
            TC  E1-ADS
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1      19
             9
         TC-TL
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1      19
            10
         TC-CV
         0.000   1.000
         0.000   1.000
         0.000   1.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
21 Do low-pressure ECC systems continue/begin to inject?
     4  E2fLPC  E2rLPC  E2aLPC  E2-LPC
     2       1       2       3       4
     4
     1      11
             2
        E1rLPC
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
     2      17      11
             1      -1
        E2nDeP nE1fLPC
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
     2      11      19
             1   +   9
        E1fLPC   TC-TL
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
       Otherwise
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
22 Do containment sprays fail before core degradation?
     4   E3fCS   E3rCS   E3aCS   E3-CS
     2       1       2       3       4
     7
     3      11      12      14
             1       1       1
        E1fLPC  E1fRHR E1fHPSW
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
     2      11      14
             2       2
        E1rLPC E1rHPSW
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
     4      12       1       1       1
             3  * (  1   +   2   +   3)
        E1-RHR       A      S1    S2/3
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
     7      21      21      14       1       1       1      19
         (   4   +   3   +   3) * (   1   +   2   +   3   +   7)
        E2-LPC  E2aLPC E1aHPSW       A      S1    S2/3      TW
         0.000   0.000   0.000   1.000
         0.000   0.000   0.010   0.990
         0.000   0.000   0.100   0.900
     2      19      19
             8   +   9
       TC-fDeP   TC-TL
         0.000   0.000   0.001   0.999
         0.000   0.000   0.010   0.990
         0.000   0.000   0.100   0.900
     1      19
            10
         TC-CV
         0.000   0.000   0.100   0.900
         0.000   0.000   0.500   0.500
         0.000   0.000   1.000   0.000
       Otherwise
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
23 Is the containment vented before core degradation?
     3 E3nVENT E3uVENT E3-VENT
     2       1       2       3
     6
     1      19
             6
            TB
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     2      19      19
             2   +   4
            AW     S1W
         0.001   0.000   0.999
         0.025   0.000   0.975
         0.100   0.000   0.900
     1      19
             7
            TW
         0.000   0.000   1.000
         0.001   0.000   0.999
         0.100   0.000   0.900
     2      19      20          TCSRX2 => 1,0,0 Otherwise 0,0,1
            10       2
         TC-CV  nPCyBk
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     2       1      20
             6       2
            TC  nPCyBk            TCSV23 & TCSAV23 => 0,0,1; OTHER TC =>
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
24 Does the RPV repressurize before core damage?
     2     ReP    nReP
     2       1       2
     7
```

402

```
     3      19      19      19
            -6      -7     -10
           nTB     nTW  nTC-CV
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1       8
             1
          SORV
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1      20
             1
         PCyBk
         0.000   1.000
         0.000   1.000
         0.000   1.000
     2      19      23
            10       3
         TC-CV E3-VENT
         0.000   1.000
         0.000   1.000
         0.000   1.000
     2      19      17
             6       2
            TB  E2-DeP   Very likely batteries fail before injection
         0.900   0.100
         0.990   0.010
         0.999   0.001
     2      17      23
             2      -3
        E2-DeP nE3-VENT   No successful venting
         1.000   0.000
         1.000   0.000
         1.000   0.000
       Otherwise (either nDeP or nReP)
         0.000   1.000
         0.000   1.000
         0.000   1.000
25 Does (do) any S/RV-tailpipe vacuum breaker(s) stick open?
     2 oSRVBkr cSRVBkr
     2       1       2
     4
     4      19       8      17      24
             5       2  (    1    +  1)
       TQUVFSB E1nSORV  E2nDeP     ReP
         0.020   0.980
         0.230   0.770
         1.000   0.000
     5      19      19       8      17      24
         (   6   +   7)      2   (   1    +  1)
            TB      TW E1nSORV  E2nDeP     ReP
       !25 1 1 !25 1 2
```

BEST COPY AVAILABLE

```
       !25 1 1 !25 1 2
       !25 1 1 !25 1 2
     1       19
              8
       TC-fDeP                 ** Cycling valve not low-low set
         0.000   1.000
         0.010   0.990
         0.100   0.900
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
26 Is there a repressurization break of low-pressure systems?
     2   RePBk  nRePBk
     2       1       2
     2
     1      24
             1
           ReP
         0.000   1.000
         0.000   1.000
         0.000   1.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
27 Does containment fail before core degradation?
     2  E3-CFP  E3nCFP
     2       1       2
     5
     2      20      26
             1  +    1
         PCyBk   RePBk
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1      23
             3
       E3-VENT
         0.000   1.000
         0.000   1.000
         0.000   1.000
     4      19      19      19      19
            -2      -4      -7     -10
           nAW    nS1W     nTW  nTC-CV
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1      19
            10
         TC-CV        TCSRX3 & TCSARX3 => 0,1; OTHER TC/R => 1,0
         1.000   0.000
         1.000   0.000
         1.000   0.000
```

BEST COPY AVAILABLE

```
      Otherwise
         1.000   0.000
         1.000   0.000
         1.000   0.000
28 What is location of early containment leakage?
      3    E3nCL  E3-CLD  E3-CLW
      2        1       2       3
      5
      2       20      26
               1   +   1
         PCyBk   RePBk
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
      2       23      23
               2   +   3
      E3uVENT E3-VENT            Vent has been opened, FP go this way
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
      2       27      19
               1      10
        E3-CFP   TC-CV           Preserve split for TCSRX2
         0.000   0.500   0.500
         0.000   0.955   0.045
         0.000   1.000   0.000
      1       27
               1
        E3-CFP
         0.000   0.500   0.500
         0.000   0.800   0.200
         0.000   1.000   0.000
      Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
29 For drywell failures, is the failure at the head seal?
      3   E3-DHF  E3-DXF  E3nDF
      2        1       2       3
      4
      2       20      26
               1   +   1
         PCyBk   RePBk
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
      2       28      19
               2      10
        E3-CLD   TC-CV           Preserve split for TCSRX2
         0.100   0.900   0.000
         0.873   0.127   0.000
         1.000   0.000   0.000
      1       28
               2
        E3-CLD
          0.100    0.900    0.000
          0.400    0.600    0.000
          0.900    0.100    0.000
       Otherwise
          0.000    0.000    1.000
          0.000    0.000    1.000
          0.000    0.000    1.000
30 What is the containment leakage level before core degradation?
     3    E3nCL   E3-CL2   E3-CL3
     2        1        2        3
     8
     2       20       26
              1   +    1
          PCyBk   RePBk                     ** V **
          0.000    0.000    1.000
          0.000    0.000    1.000
          0.000    0.000    1.000
     1       23
              2
       E3uVENT
          0.000    1.000    0.000
          0.000    1.000    0.000
          0.000    1.000    0.000
     1       23
              3
       E3-VENT
          0.000    0.000    1.000
          0.000    0.000    1.000
          0.000    0.000    1.000
     1       19
             10
          TC-CV                        TCSRX2 & TCSARX2 => 0,1,0; OTHER TC/R
          0.000    1.000    0.000
          0.000    1.000    0.000
          0.000    1.000    0.000
     2        1       27
              6        1
             TC   E3-CFP
          0.000    0.000    1.000
          0.000    0.000    1.000
          0.000    0.000    1.000
     2       28       29
          (   3    +   2)
         E3-CLW   E3-DKF
          0.000    0.900    0.100
          0.000    0.100    0.900
          0.000    0.000    1.000
     1       29
              1
         E3-DHF
          0.000    1.000    0.000
          0.000    1.000    0.000
          0.000    1.000    0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
31 Is the suppression pool drained before CD?
     3  E4-SPD  E4pSPD  E4nSPD
     2       1       2       3
     2
     3      27      28      30
             1       3       3
        E3-CFP  E3-CLW  E4-CL3
         0.000   0.010   0.990
         0.040   0.360   0.600
         0.100   0.800   0.100
       Otherwise
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
32 Does ac power remain lost during core degradation?
     2   E4fAC   E4-AC
     2       1       2
     5
     2      19      24
             6       1
            TB     ReP  No DC @ 7 hrs -- Recovery by 11.6 hr
         0.179   0.821
         0.465   0.535
         0.596   0.404
     1      19
             6
            TB          TB but batteries survive
         0.139   0.861
         0.363   0.637
         0.465   0.535
     2       5       6
             1       2
            SB   E1-DC  Recovery @ 1.8 hr given none @ 35min
         0.533   0.467
         0.688   0.312
         0.812   0.188
     2       5       6
             1       1
            SB   E1fDC  Recovery @ 1.8 hr given none @ 35min
         0.630   0.370
         0.812   0.188
         0.959   0.041
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
33 What is the status of CRD?
     3 E34fCRD E34rCRD E34-CRD
     2       1       2       3
     5
     7      23      30      10      10      32      23       5
         (   2   +   3)  (   3  + (  2   *   2))  (   1   +   2)
       E3uVENT  E3-CL3  E1-CRD  E1rCRD   E4-AC E3nVENT     nSB
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     4      30      10      23       5
             3       2   (   1   +   2)
        E3-CL3  E1rCRD E3nVENT     nSB
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     3      10      10      32
         (   3  + (  2   *   2))
        E1-CRD  E1rCRD   E4-AC
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1      10
             2
        E1rCRD
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
34 What is the status of HPSW?
     3 E34fHSW E34rHSW E34aHSW
     2       1       2       3
     5
     7      23      30      14      14      32      23       5
         (   2   +   3)  (   3  + (  2   *   2))  (   1   +   2)
       E3uVENT  E3-CL3 E1aHPSW E1rHPSW   E4-AC E3nVENT     nSB
         0.001   0.000   0.999
         0.100   0.000   0.900
         0.500   0.000   0.500
     4      30      14      23       5
             3       2   (   1   +   2)
        E3-CL3 E1rHPSW E3nVENT     nSB
         0.001   0.999   0.000
         0.100   0.900   0.000
         0.500   0.500   0.000
     3      14      14      32
         (   3  + (  2   *   2))
       E1aHPSW E1rHPSW   E4-AC
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1      14
             2
       E1rHPSW
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
35 Does the core melt?
     2    MELT   nMELT
     2       1       2
     9
     3      19      19      13
          (  2   +   4)      3
            AW     S1W E1aCOND
         0.000   1.000
         0.001   0.999
         0.100   0.900
     3      19      19      34
          (  2   +   4)      3
            AW     S1W E34aHSW
         0.000   1.000
         0.001   0.999
         0.100   0.900
     3      19      19      33
          (  2   +   4)      3
            AW     S1W E34-CRD
         0.001   0.999
         0.100   0.900
         0.500   0.500
     2      19      33
             7       3
            TW E34-CRD
         0.000   1.000
         0.000   1.000
         0.000   1.000
     6      19      13      17      24      26      30
             7       3       2   (   2   +   1   +   3))
            TW E1aCOND  E2-DeP    nReP   RePBk  E3-CL3
         0.000   1.000
         0.001   0.999
         0.100   0.900
     6      19      34      17      24      26      30
             7       3       2   (   2   +   1   +   3))
            TW E34aHSW  E2-DeP    nReP   RePBk  E3-CL3
         0.000   1.000
         0.001   0.999
         0.100   0.900
     6      19      13      17      24      26      30
            10       3       2   (   2   +   1   +   3))
         TC-CV E1aCOND  E2-DeP    nReP   RePBk  E3-CL3
         0.000   1.000
         0.001   0.999
         0.100   0.900
     6      19      34      17      24      26      30
            10       3       2   (   2   +   1   +   3))
         TC-CV E34aHSW  E2-DeP    nReP   RePBk  E3-CL3
         0.001   0.999
         0.010   0.990
         0.100   0.900
       Otherwise
         1.000   0.000
         1.000   0.000
         1.000   0.000
36 What is containment pressure before core degradation?
     1  PBase1
     4
    12
     2      19      22
             1       4
            AE   E3-CS
         1.000
         1.000
         1.000
     1
     1    17.0
          30.0                        BMI-2104
          43.0
     1      19
             1
            AE
         1.000
         1.000
         1.000
     1
     1    30.0
          35.0                        BMI-2104
          52.0
     2      19      12
             3       3
           S1E  E1-RHR
         1.000
         1.000
         1.000
     1
     1    15.0
          17.0
          19.0
     1      19
             3
           S1E
         1.000
         1.000
         1.000
     1
     1    30.0
          40.0
          52.0
     3      19       9      22
             8      -3       4
```

```
  TC-fDeP nE1-HPC   E4-CS
    1.000
    1.000
    1.000
1
1    15.0
     15.0
     17.0
3       19      19       9
        5  + (   8      -3)
  TQUV/FTB TC-fDeP nE1-HPC
    1.000
    1.000
    1.000
1
1    16.0
     18.0
     20.0
3       19      19       9
        6  + (   8       3)
       TB TC-fDeP  E1-HPC
    1.000
    1.000
    1.000
1
1    22.0
     31.0
     40.0
1       20
         1
    PCyBk
    1.000
    1.000
    1.000
1
1    20.0
     60.0
    100.0
1       26
         1
    RePBk
    1.000
    1.000
    1.000
1
1   100.0                          CHECK ReP THRESHOLD
    110.0
    120.0
2        1      30
         6       1
        TC   E3nCL
    1.000
    1.000
    1.000
```

```
     1
     1   105.0
         125.0
         130.0
     1      30
             2
        E3-CL2
         1.000
         1.000
         1.000
     1
     1   160.0
         130.0
         100.0
       Otherwise
         1.000
         1.000
         1.000
     1
     1    20.0
          20.0
          20.0
37 Will the SP flash following containment vent or rupture?
     2   SatSP  SubcSP
     2       1       2
     5
     6       9      12      19      19      24       8
         (   3   *  -3   *  -6) + (  6   *   2   *   2)
        E1-HPC  E1nRHR     nTB      TB    nReP E1nSORV
         1.000   0.000
         1.000   0.000
         1.000   0.000
     1      19
             6
            TB           Only subcooled if vessel over-filled
         0.900   0.100
         0.990   0.010
         1.000   0.000
     2      21      12
             4      -3
        E2-LPC  E1nRHR
         1.000   0.000
         1.000   0.000
         1.000   0.000
     3       1       9      21
             6   (   3   +   4)
            TC  E1-HPC  E2-LPC
         1.000   0.000
         1.000   0.000
         1.000   0.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
```

```
38 Is the RPV depressurized during core degradation?
     2  E4nDeP  E4-DeP
     2       1       2
     7
     2      20      26
         (   1   +   1)
        PCycBk   RePBk
         0.000   1.000
         0.000   1.000
         0.000   1.000
     6      24       5      30      23       5      32
             1   ((  2     ( 3   + -1))  + ( 1   *   2))
           ReP     nSB  E3-CL3 E3-VENT      SB   E4-AC
         0.000   1.000
         0.000   1.000
         0.000   1.000
     2      17      19
             1       8
        E2nDeP TC-fDeP
         0.100   0.900
         0.500   0.500
         0.900   0.100
     7      17       5      32      21      37      30      23
             1       1       2      -1 * (   2   + ( 1       1))
        E2nDeP      SB   E4-AC nE2fLPC   SubSP   E3nCL E3nVENT
         0.000   1.000
         0.001   0.999
         0.010   0.990
     3      17       5      32
             1       1       2
        E2nDeP      SB   E4-AC   ADS unavailable, no prior manual
         0.900   0.100
         0.500   0.500
         1.000   0.000
     2      17      24
             1   +   1
        E2nDeP     ReP
         1.000   0.000
         1.000   0.000
         1.000   0.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
39 Is there injection during core degradation?
     3 E4-LPIn E4-CDIn   E4nIn
     2       1       2       3
     5
     6      21      37      31      30      32      38
         (  -1 * (   2      -1   +   1))       2   *   2
       nE2fLPC   SubSP nE4-SPD   E3nCL   E4-AC  E5-DeP
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     3      13      32      38
            -1       2       2
       nE1fCOND  E4-AC  E5-DeP
         1.000   0.000   0.000
         0.999   0.000   0.001
         0.990   0.000   0.010
     2      34      38
             3       2
       E5-HPSW  E5-DeP
         0.999   0.000   0.001
         0.900   0.000   0.100
         0.500   0.000   0.500
     1      33
             3
       E34-CRD
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
40 What is the status of containment sprays?
     3   E4fCS   E4rCS   E4-CS
     2       1       2       3
     5
     1      22
             1
         E3fCS
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     2      22      32
             2       1
         E3rCS   E4fAC
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     4      22      37      31      30
             4    (  2      -1   +   1)
         E4-CS  SubcSP nE4-SPD   E3nCL
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     5      21      32      30      37      31
            -1       2   (   1  + (  2      -1))
       nE2fLPC   E4-AC   E3nCL  SubcSP nE4-SPD
         0.000   0.000   1.000
         0.100   0.000   0.900
         0.500   0.000   0.500
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
```

```
41 What is the level of flow to the drywell during CD?
     3  E4nFDW E4-FDW2 E4-FDW3
     2       1       2       3
     6
     4       1       1       1       38
             1   +   2   + ( 3        2)
             A      S1    S2/3  E4-DeP   ** to DW and CLD   **
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     2      20      26
             1   +   1
        PCycBk   RePBk          ** V -  break outside cont. **
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     2       1      38
             3       1
          S2/3  E4nDeP
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     2      25      38
             1       1
       oSRVbkr  E4nDeP
         0.990   0.010   0.000
         0.800   0.200   0.000
         0.000   1.000   0.000
     1      25
             1
       oSRVbkr
         0.000 !41 4 1 !41 4 2
         0.000 !41 4 1 !41 4 2
         0.000 !41 4 1 !41 4 2
       Otherwise                       ** everything through SP **
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
42 What is the H2 generation and pressure rise during CD?
     3   H2Low   H2Med    H2Hi         DP-H2bVB
     4       1       2       3
    10
     4      20      26      23       30
             1   +   1   +  -1   +    3
         PCyBk   RePBk E3-VENT  E3-CL3
         1.000   0.000   0.000
         0.000   1.000   0.000
         0.000   0.000   1.000
     1
     2     0.0     0.0     0.0
           0.0     0.0     0.0
           0.0     0.0     0.0
     2      39      41
             1       3
     E4-LPI E4-FWD3                             Injection during melt
      0.500   0.500   0.000
      0.000   0.500   0.500
      0.000   0.000   1.000
1
2      17.0    46.0    79.0   Med = 11 psi + TB, Hi=2*TB + 11psi
       17.0    46.0    79.0
       17.0    46.0    79.0
1        39
          1
     E4-LPI                   Injection during melt
      0.500   0.500   0.000
      0.000   0.500   0.500
      0.000   0.000   1.000
1
2      13.0    33.0    55.0   Med = 11 psi + TB, Hi=2*TB + 11psi
       13.0    33.0    55.0
       13.0    33.0    55.0
2         1       1
          1   +   2
          A      S1                **LOCA, to drywell **
      1.000   0.000   0.000
      0.000   1.000   0.000
      0.000   0.000   1.000
1
2      17.0    45.0   106.0
       17.0    45.0   106.0
       17.0    45.0   106.0
1        33
          3
    E34-CRD                  **Transient with CRD Flow**
      1.000   0.000   0.000
      0.000   1.000   0.000
      0.000   0.000   1.000
1
2      30.0    51.0    70.0
       30.0    51.0    70.0
       30.0    51.0    70.0
1        41
          3
    E4-FWD3                  **Transient with oSRVBkr**
      1.000   0.000   0.000
      0.000   1.000   0.000
      0.000   0.000   1.000
1
2      17.0    34.0    68.0
       17.0    34.0    68.0
       17.0    34.0    68.0
3        17      24      38
      (   1   +   1)      2
     E2nDeP     ReP  E4-DeP
      1.000   0.000   0.000
      0.000   0.900   0.100
      0.000   0.000   1.000
     1
     2     22.0    38.0    76.0
           22.0    38.0    76.0
           22.0    38.0    76.0
     1       19
              6
             TB
          1.000   0.000   0.000
          0.000   1.000   0.000
          0.000   0.000   1.000
     1
     2     13.0    22.0    37.0
           13.0    22.0    37.0
           13.0    22.0    37.0
     2       17      24
              1  +    1
         E2nDeP     ReP        ** High Pressure
          1.000   0.000   0.000
          0.000   1.000   0.000
          0.000   0.000   1.000
     1
     2     11.0    24.0    48.0
           11.0    24.0    48.0
           11.0    24.0    48.0
       Otherwise               ** Low Pressure **
          1.000   0.000   0.000
          0.000   1.000   0.000
          0.000   0.000   1.000
     1
     2     22.0    38.0    76.0
           22.0    38.0    76.0
           22.0    38.0    76.0
43 Does at least one drywell vacuum breaker stick open?
     2   E4-VBo  E4nVBo
     2        1       2
     7
     5        1       1      20      26      28
              1  +    2  +    1  +    1  +    3
              A      S1   PCyBk   RePBk  E3-CLW
          0.000   1.000
          0.000   1.000
          0.000   1.000
     2       19      19
              7  +   10
             TW   TC-CV
          0.004   0.996
          0.004   0.996
          0.040   0.960
     4       17      24      25       1
              2       2  (    1   +    3)
         E2-DeP    nReP oSRVBkr    S2/3       VBkrs open may, but gradua
          0.000   1.000
          0.002   0.998
          0.020   0.980
     2      17      24
             2       2
        E2-DeP    nReP          VBkrs open, but gradual h2 release
         0.001   0.999
         0.002   0.998
         0.040   0.960
     1      42
             3
          H2Hi                  VBkrs open, rapid H2 release
         0.007   0.993
         0.014   0.986
         0.300   0.700
     1      42
             2
         H2Med                  VBkrs open, rapid H2 release
         0.000   1.000
         0.005   0.995
         0.200   0.800
       Otherwise       >H2Low VBkrs open, rapid H2 release
         0.000   1.000
         0.001   0.999
         0.030   0.970
44 Does H2 discharge lead to SP bypass?
     2  H2-SPB  H2nSPB
     2       1       2
     4
     2      17      24
             2       2
        E2-DeP    nReP
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1      42
             3
          H2Hi
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1      42
             2
         H2Med
         0.000   1.000
         0.000   1.000
         0.000   1.000
       Otherwise       >H2Low
         0.000   1.000
         0.000   1.000
         0.000   1.000
45 Does H2 discharge to SP induce WW failure?
     3  E4nHWF E4-HWF2 E4-HWF3
     2       1       2       3
     3
     2      17      24
             2   *   2
        E2-DeP    nReP
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     1      44
             1
        H2-SPB
         0.980   0.000   0.020
         0.900   0.000   0.100
         0.500   0.000   0.500
       Otherwise        > No SP Bypass
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
46 Is the vent threshold reached during core degradation?
     2  E4-Vth  E4nVth
     6       1       2
     2
     1      45
            -3
       E4nHWF3
     2       1       2
        PBase1   DP-H2
           AND
        THRESH       1    75.0
                          75.0
                          75.0
                       VThresh
       Otherwise
     1       1
        PBase1
           AND
        THRESH       1   999.0
                         999.0
                         999.0
                       VThresh
47 Does containment venting not occur during core degradation?
     2 E4nVENT E4-VENT
     2       1       2
     8
     1      23
             2
       E3uVENT
         0.000   1.000
         0.000   1.000
         0.000   1.000
     3      19      30      46
             7       1       1
            TW   E3nCL  E4-Vth If Op didn't vent yet, he won't
         1.000   0.000
         1.000   0.000
         1.000   0.000
     1      30
             2
```

419

```
        E3-CL2
         0.050   0.950
         0.500   0.500
         0.000   1.000
     2       5      46
             2       1
           nSB  E4-Vth   INEL TC2 evaluation
         0.050   0.950
         0.230   0.770
         1.000   0.000
     6       5      46      32      12      40      42
             1       1       2   (   1   +   1   +   3)
            SB  E4-Vth   E4-AC  E1fRHR   E4fCS    H2Hi
         0.001   0.999
         0.010   0.990
         0.100   0.900
     3       5      46      32
             1       1       2
            SB  E4-Vth   E4-AC
         1.000   0.000
         0.900   0.100
         0.500   0.500
     2      19      46
             6       1
            TB  E4-Vth
         0.500   0.500
         0.900   0.100
         1.000   0.000
       Otherwise
         1.000   0.000
         1.000   0.000
         1.000   0.000
48 Does the RPV repressurize during core degradation?
     2  E4-ReP  E4nReP
     6       1       2
     2
     6      47       1       1       8      45      38
             1      -1      -2       2      -3       2
       E4nVENT      nA     nS1   nSORV E4nHWF3  E4-DeP
     2       1       2
        Pbase1   DP-H2
        'AND'
       'THRESH'      1   130.0
                         120.0
                         110.0
                       RPThresh
       Otherwise
     1       1
        Pbase1
        'AND'
       'THRESH'      1   999.0
                         999.0
                         999.0
                       RPThresh
```

420

```
49 Is there a repressurization break during core degradation?
     2 E4-RePB E4nRePB
     2       1       2
     2
     1      48
             1
        E4-ReP
         0.000   1.000
         0.000   1.000
         0.000   1.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
50 Does the containment fail by pressure during core degradation?
     2  E4-CFP  E4nCFP
     6       1       2
     2
     5      30      47      49      45      19
            -3       1       2      -3     -10
       nE3-CL3 E4nVENT E4nRePB E4nHWF3  nTC-CV
     2       1       2
        Pbase1   DP-H2
           AND
        NORMAL       2   175.0    15.0
                         132.0     5.0
                         132.0    15.0
                CFailP    Mean   Sigma
       Otherwise
     1       1
        Pbase1
           AND
        THRESH       1   999.0
                         999.0
                         999.0
               Prevent containment failure
51 What is location of containment leakage prior to vessel breach?
     3   E4nCL  E4-CLD  E4-CLW
     2       1       2       3
     4
     4      28      30      47      49
        (    2   (   3  +    1))  +   1
        E3-CLD  E3-CL3 E4nVENT E4-RePB
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     3      28      45      47
             3   +  -1  +    2
        E3-CLW  E4-HWF E4-VENT
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1      50
             1
       E4-CFP
      !28 4 1 !28 4 2 !28 4 3
      !28 4 1 !28 4 2 !28 4 3
      !28 4 1 !28 4 2 !28 4 3
      Otherwise
        1.000   0.000   0.000
        1.000   0.000   0.000
        1.000   0.000   0.000
52 For drywell failures, is the failure at the head seal?
   3   E4-DHF  E4-DKF   E4nDF
   2        1       2       3
   4
   2       29      47
            1       1
       E3-DHF E4nVENT
        1.000   0.000   0.000
        1.000   0.000   0.000
        1.000   0.000   0.000
   3       29      30      47
            2   (   3   +   1)
       E3-DKF  E3-CL3 E4nVENT
        0.000   1.000   0.000
        0.000   1.000   0.000
        0.000   1.000   0.000
   1       51
            2
       E4-CLD
      !29 3 1 !29 3 2 !29 3 3
      !29 3 1 !29 3 2 !29 3 3
      !29 3 1 !29 3 2 !29 3 3
      Otherwise
        0.000   0.000   1.000
        0.000   0.000   1.000
        0.000   0.000   1.000
53 What is the level of containment leakage before VB?
   3    E4nCL  E4-CL2  E4-CL3
   2        1       2       3
   7
   4       30      19      29      45
            3 + (  10      -1)  +   3
       E3-CL3   TC-CV nE3-DHF E4-HWF3
        0.000   0.000   1.000
        0.000   0.000   1.000
        0.000   0.000   1.000
   1       49
            1
      E4-RePBk
        0.000   0.000   1.000
        0.000   0.000   1.000
        0.000   0.000   1.000
   1       47
            2
      E4-VENT
        0.000   0.000   1.000
```

```
         0.000   0.000   1.000
         0.000   0.000   1.000
     3      50      52      51
             1   (   2   +   3)
        E4-CFP  E4-DKF  E4-CLW
       !30 6 1 !30 6 2 !30 6 3
       !30 6 1 !30 6 2 !30 6 3
       !30 6 1 !30 6 2 !30 6 3
     1      52
             1
        E4-DHF
       !30 7 1 !30 7 2 !30 7 3
       !30 7 1 !30 7 2 !30 7 3
       !30 7 1 !30 7 2 !30 7 3
     2      30      45
             2   +   2
        E3-CL2 E4-HWF2
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
54 What is the status of CRD?
     3  E5fCRD  E5rCRD  E5-CRD
     2       1       2       3
     5
     6      50      53      33      23      32      5
             1       3       3   (   1   +   2  +   2)
        E4-CFP  E4-CL3 E34-CRD E3nVENT   E4-AC    nSB
       !33 1 1 !33 1 2 !33 1 3
       !33 1 1 !33 1 2 !33 1 3
       !33 1 1 !33 1 2 !33 1 3
     6      50      53      33      23      32      5
             1       3       2   (   1   +   2  +   2)
        E4-CFP  E4-CL3 E34rCRD E3nVENT   E4-AC    nSB
       !33 2 1 !33 2 2 !33 2 3
       !33 2 1 !33 2 2 !33 2 3
       !33 2 1 !33 2 2 !33 2 3
     1      33
             3
       E34-CRD
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1      33
             2
       E34rCRD
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         1.000   0.000   0.000
```

423

```
         1.000   0.000   0.000
         1.000   0.000   0.000
55 What is the status of HPSW?
     4 E5fHPSW E5rHPSW E4aHPSW E5-HPSW
     2       1       2       3       4
     8
     6      39      13      21      37      30      31
             1   (   1   * (   1 + (   1      -1   +   1)))
        E4-LPI E1fCOND  E2fLPC   SatSP   E3-CL  E4-SPD
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
     4       5      23      47      34
             1   (   3   +   2)      2
            SB E3-VENT E4-VENT E34rHSW
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
     3       5      23      34
             1       3       3
            SB E3-VENT E34aHSW
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
     2      53      34
             3       3
        E4-CL3 E34aHSW
       !34 1 1 !34 1 2 !34 1 3   0.000
       !34 1 1 !34 1 2 !34 1 3   0.000
       !34 1 1 !34 1 2 !34 1 3   0.000
     2      53      34
             3       2
        E4-CL3 E34rHSW
       !34 2 1 !34 2 2 !34 2 3   0.000
       !34 2 1 !34 2 2 !34 2 3   0.000
       !34 2 1 !34 2 2 !34 2 3   0.000
     1      34
             3
       E34aHSW
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
     1      34
             2
       E34rHSW
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
       Otherwise
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
56 Is MSIV leakage control or condenser vacuum maintained?
     2 MSIVnLC MSIV-LC
     2       1       2
     2
     1      32
             2
         E4-AC
         0.000   1.000
         0.001   0.999
         0.100   0.900
       Otherwise
         1.000   0.000
         1.000   0.000
         1.000   0.000
57 Is the suppression pool drained before VB?
     3  E5-SPD  E5pSPD  E5nSPD
     2       1       2       3
     5
     1      31
             1
        E4-SPD
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     1      31
             2
        E4pSPD
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     6      50      53      51      27      28      30
         (   1       3       3) + (   1       3       2)
        E4-CFP  E4-CL3  E4-CLW  E3-CFP  E3-CLW  E4-CL2
       !31 1 1 !31 1 2 !31 1 3
       !31 1 1 !31 1 2 !31 1 3
       !31 1 1 !31 1 2 !31 1 3
     1      45
             3
       E4-HWF3
         0.000   0.010   0.990
         0.010   0.090   0.900
         0.050   0.450   0.500
       Otherwise
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
58 What is the level of suppression-pool bypass at VB?
     3  E5nSPB E5-SPB2 E5-SPB3
     2       1       2       3
     5
     3      51      53      57
             2       3   +  -3
        E4-CLD  E4-CL3  E5-SPD
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
```

```
      3       20       26       49
               1    +   1   +    1
         PCycBk    RePBk  E4-RePB ** V -  break outside cont. **
          0.000    0.000    1.000
          0.000    0.000    1.000
          0.000    0.000    1.000
      2       43       44
               1    +   1
         E4-VBo   H2-SPB
          0.000    0.000    1.000
          0.000    0.000    1.000
          0.000    0.000    1.000
      2       51       53
               2        2
         E4-CLD   E4-CL2
          0.000    1.000    0.000
          0.000    1.000    0.000
          0.000    1.000    0.000
        Otherwise                          ** everything through SP **
          1.000    0.000    0.000
          1.000    0.000    0.000
          1.000    0.000    0.000
 59 What is level of RB breach/bypass before VB without burn?
      4  E5nRBB E5-RBB2 E5-RBB3 E5-RBB4
      2        1        2        3        4
      7
      3       20       26       49
               1    +   1   +    1
         PCycBk    RePBk  E4-RePB          ** V **
          0.000    1.000    0.000    0.000
          0.000    1.000    0.000    0.000
          0.000    1.000    0.000    0.000
      4        5       23       47       32
               1    (  -1   + (  2    *   1))
              SB E3-Vent  E4-Vent     E4fAC
          0.000    0.000    0.000    1.000
          0.000    0.000    0.000    1.000
          0.000    0.000    0.000    1.000
      1       52
               1
         E4-DHF
          0.000    0.000    1.000    0.000
          0.000    0.000    1.000    0.000
          0.000    0.000    1.000    0.000
      2       23       47
               3    +   2
        E3-Vent  E4-Vent
          0.000    1.000    0.000    0.000
          0.000    1.000    0.000    0.000
          0.000    1.000    0.000    0.000
      1       53
               3
         E4-CL3
          0.000    1.000    0.000    0.000
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
     1      53
             2
        E4-CL2
         1.000   0.000   0.000   0.000
         0.500   0.500   0.000   0.000
         0.900   0.100   0.000   0.000
       Otherwise
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
60 Are the fire sprays actuated before VB without burn?
     2  EFireS nEFireS
     2       1       2
     2
     5      58      37      59      59      53
         (  -1   +   1)  (   1   +   2)      -1
        E5-SPB   SatSP  E5nRBB E5-RBB2    E4-CL
         1.000   0.000
         1.000   0.000
         1.000   0.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
61 Does SGTS fail before VB without burn?
     3 E5fSGTS E5rSGTS E5-SGTS
     2       1       2       3
     4
     4      60      32      23      47
             2       2       1       1
       nEFireS   E4-AC E3nVent E4nVent ** no venting **
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     2      60      32
             2       2
       nEFireS   E4-AC                 ** vented to SGTS **
         0.100   0.000   0.900
         0.500   0.000   0.500
         0.900   0.000   0.100
     2      60      32
             2       1
       nEFireS   E4fAC
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise                       ** EFireS **
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
62 Does hydrogen burn in RB before VB?
     2   HBbVB  nHBbVB
     2       1       2
     8
     3      20      26      49
             1   +   1   +   1
        PCycBk   RePBk E4-RePB
         1.000   0.000
         1.000   0.000
         1.000   0.000
     1      59
             4
       E5-RBB4
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1      42
             1
         LowH2
         0.000   1.000
         0.000   1.000
         0.000   1.000
     2      53      61
             2       3
        E4-CL2 E5-SGTS
         0.000   1.000
         0.100   0.900
         0.500   0.500
     1      59
             3
       E5-RBB3
         0.500   0.500
         0.999   0.001
         1.000   0.000
     2      53      61
             3       3
        E4-CL3 E5-SGTS
         0.500   0.500
         0.999   0.001
         1.000   0.000
     2      53      61
            -1      -3
         E4-CL E5nSGTS
         0.999   0.001
         1.000   0.000
         1.000   0.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
63 What is level of reactor building (RB) breach/bypass before VB?
     4   nERBB   ERBB2   ERBB3   ERBB4
     2       1       2       3       4
     6
     1      59
             4
```

BEST COPY AVAILABLE

```
       E5-RBB4
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
     3      52      59      62
         (   1   +   3)       1
        E4-DHF E5-RBB3   HBbVB
         0.000   0.000   0.500   0.500
         0.000   0.000   0.100   0.900
         0.000   0.000   0.000   1.000
     3      59      59      62
         (   1   +   2)       1
       E5-RBB1 E5-RBB2   HBbVB
         0.000   0.980   0.000   0.020
         0.000   0.780   0.000   0.220
         0.000   0.000   0.000   1.000
     1      59
             3
       E5-RBB3
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
     1      59
             2
       E5-RBB2
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
       Otherwise
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
64 Are fire systems actuated before VB?
     2  EFireS nEFireS
     2       1       2
     2
     3      60      62      52
             2       2   +   1
       nEFireS  nHBbVB  E4-DHF
         0.000   1.000
         0.000   1.000
         0.000   1.000
       Otherwise
         1.000   0.000
         1.000   0.000
         1.000   0.000
65 Does SGTS operate until VB?
     3  EfSGTS  ErSGTS  E-SGTS
     2       1       2       3
     3
     3      61      64      62
             3       2       2
       E5-SGTS nEFireS  nHBbVB
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     3      61      64      62
             2       2       2
       E5rSGTS nEFireS  nHBbVB
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
66 What is containment pressure after CF or vent during meltdown?
     1  PBase2
     4
     2
     1      53
             3
        E4-CL3
         1.000
         1.000
         1.000
     1
     3    20.0
          20.0
          20.0
       Otherwise       ** CL Level 2 Only**
         1.000
         1.000
         1.000
     1
     3   170.0
         132.0
         100.0
67 Is there injection to the RPV up to the time of vessel breach?
     3 E5-LPIn E5-HPIn  nE5-In
     2       1       2       3
     6
     6      39      37      57      53      48      53
             1 ((    2      -1   +   1)  (   2   +   3))
       E4-LPIn   SubSP nE5-SPD   E4nCL  E4nReP  E4-CL3
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     7      39      48      53      21      37      30      31
             1   (   2   +   3)  (   1 + (   1      -1   +   1))
       E4-LPIn  E4nReP  E4-CL3  E2fLPC   SatSP   E3-CL  E4-SPD
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     6      39      48      53      13      55      54
             1   (   2   +   3)  (  -1   +   3)       3
       E4-LPIn  E4nReP  E4-CL3 nE1fCOND E5aHPSW  E5-CRD
         1.000   0.000   0.000
         0.990   0.010   0.000
         0.900   0.100   0.000
     5      39      48      53      13      55
             1   (   2   +   3)  (  -1   +   3)
       E4-LPIn  E4nReP  E4-CL3 nE1fCOND E5aHPSW
         1.000   0.000   0.000
         0.990   0.000   0.010
         0.900   0.000   0.100
     1      54
             3
        E5-CRD
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
68 What is the nature of meltdown?
     3   Slump    Flow   nMelt
     2       1       2       3
     4
     1      35
             2
         nMelt
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1      67
             1
       E5-LPIn
         0.000   1.000   0.000
         0.100   0.900   0.000
         0.500   0.500   0.000
     1      67
             2
       E5-HPIn
         0.000   1.000   0.000
         0.100   0.900   0.000
         0.500   0.500   0.000
       Otherwise                 ** no injection **
         0.001   0.999   0.000
         0.100   0.900   0.000
         0.500   0.500   0.000
69 What is the mode of vessel breach?
     4   Alpha   SE-BH   Mthru nBreach
     2       1       2       3       4
     5
     1      35
             2
         nMelt
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
         0.000   0.000   0.000   1.000
     1      68
             1
         Slump
        0.0000  0.0000  1.0000  0.0000
        0.0001  0.0000  0.9999  0.0000
        0.0100  0.2100  0.7800  0.0000
     1      67
             1
       E5-LPIn
         0.000   0.000   0.100   0.900
         0.000   0.000   0.500   0.500
         0.000   0.000   0.900   0.100
     2      67      19
             2       6
       E5-HPIn      TB
         0.000   0.000   0.900   0.100
         0.000   0.000   0.999   0.001
         0.000   0.000   1.000   0.000
       Otherwise            =>    Flow ** no injection **
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
70 Is the RPV blown down before significant melt ejection?
     2 nEBlowD  EBlowD
     2       1       2
     3
     3      38      48      53
             2   (   2   +   3)
        E5-Dep  E4nReP  E4-CL3
         0.000   1.000
         0.000   1.000
         0.000   1.000
     2      69      68
             3       2
         Mthru    Flow
         0.000   1.000
         0.400   0.600
         0.900   0.100
       Otherwise
         1.000   0.000
         1.000   0.000
         1.000   0.000
71 Does direct heating occur?
     2      DH     nDH
     2       1       2
     2
     2      68      70
             1       1
         Slump nEBlowD         ** requries slump and HiP **
         0.500   0.500
         0.900   0.100
         1.000   0.000
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
72 Does an ex-vessel steam explosion occur?
     2    ExSE   nExSE
     2       1       2
     2
     2      68      69
             1       3
         Slump   Mthru          ** requires slump **
         0.000   1.000
         0.010   0.990
         0.100   0.900
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
73 What is pressure rise due to blowdown (effective VS and RCdry)?
     1   DPblow
     4       1
     3
     1      71
             1
            DH
         1.000
         1.000
         1.000
     1
     4    30.0
          96.0
         116.0
     1      38
             1
        E5nDEP
         1.000
         1.000
         1.000
     1
     4    10.0
          76.0
          96.0
       Otherwise                ** LoP **
         1.000
         1.000
         1.000
     1
     4     0.0
           0.0
           0.0
74 What is the pressure rise due to debris quenching?
     2   RCwet   RCdry
     4       1       2
     3
     3       1       1       1
             1   +   2   +   3
```

433

```
             A      S1    S2/3 ** loca **
         1.000   0.000
         1.000   0.000
         1.000   0.000
     1
     5     7.0     5.0
          15.0    10.0
          45.0    30.0
     3      22      40      67
             4   +   3   +   1
         E3-CS   E4-CS E4-LPIn
         1.000   0.000
         1.000   0.000
         1.000   0.000
     1
     5    15.0     5.0
          30.0    10.0
          45.0    30.0
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
     1
     5     7.0     5.0
          15.0    10.0
          40.0    30.0
75 Is SP bypassed or vents not cleared at VB (effect. VS)? -- DPfVS
     3    Vclr   nVclr  Bypass
     4       1       2       3
    15
     4      71      72      37      58
             2       1       1      -3
           nDH    ExSE   SatSP nE5SPB3
         0.900   0.100   0.000
         0.500   0.500   0.000
         0.500   0.500   0.000
     1
     6    25.0    70.0    40.0
          50.0   125.0    70.0
          75.0   180.0   100.0
     3      71      72      58
             2       1      -3
           nDH    ExSE nE5SPB3
         0.900   0.100   0.000
         0.500   0.500   0.000
         0.500   0.500   0.000
     1
     6    20.0    50.0    30.0
          45.0   105.0    60.0
          70.0   160.0    90.0
     4      71      72      37      58
             1       2       1      -3
            DH   nExSE   SatSP nE5SPB3
         0.980   0.020   0.000
       0.840     0.160     0.000
       0.500     0.500     0.000
1
6        0.0     999.0     999.0
         0.0     999.0     999.0
         0.0     999.0     999.0
3         71        72        58
           1         2        -3
          DH     nExSE   nE5SPB3
       0.999     0.001     0.000
       0.840     0.160     0.000
       0.600     0.400     0.000
1
6        0.0     999.0     999.0
         0.0     999.0     999.0
         0.0     999.0     999.0
4         71        72        37        58
           1         1         1        -3
          DH      ExSE     SatSP   nE5SPB3
       0.900     0.100     0.000
       0.500     0.500     0.000
       0.500     0.500     0.000
1
6      999.0     999.0     999.0
       999.0     999.0     999.0
       999.0     999.0     999.0
3         71        72        58
           1         1        -3
          DH      ExSE   nE5SPB3
       0.900     0.100     0.000
       0.500     0.500     0.000
       0.500     0.500     0.000
1
6      999.0     999.0     999.0
       999.0     999.0     999.0
       999.0     999.0     999.0
3         74        37        58
           1         1        -3
       RCwet     SatSP   nE5SPB3
       1.000     0.000     0.000
       1.000     0.000     0.000
       1.000     0.000     0.000
1
6        3.0       0.0       0.0
         6.0       0.0       0.0
        12.0       0.0       0.0
1         58
          -3
     nE5SPB3
       1.000     0.000     0.000
       1.000     0.000     0.000
       1.000     0.000     0.000
1
6        0.0       0.0       0.0
```

```
          0.0       0.0       0.0
          0.0       0.0       0.0
1         53
           3
      E4-CL3          ** Bypass but previous failure **
       0.000     0.000     1.000
       0.000     0.000     1.000
       0.000     0.000     1.000
1
6         0.0       0.0       0.0
          0.0       0.0       0.0
          0.0       0.0       0.0
2         71        74
           1         2
          DH     RCDry
       0.000     0.000     1.000
       0.000     0.000     1.000
       0.000     0.000     1.000
1
6         0.0       0.0      60.0
          0.0       0.0      70.0
          0.0       0.0     120.0
1         71
           1
          DH
       0.000     0.000     1.000
       0.000     0.000     1.000
       0.000     0.000     1.000
1
6         0.0       0.0      75.0
          0.0       0.0     105.0
          0.0       0.0     180.0
3         74        38        58
           1         1         3
       RCwet    E5nDEP   E5-SPB3
       0.000     0.000     1.000
       0.000     0.000     1.000
       0.000     0.000     1.000
1
6         0.0       0.0      80.0
          0.0       0.0     120.0
          0.0       0.0     180.0
3         74        38        58
           1         2         3
       RCwet    E5-DEP   E5-SPB3
       0.000     0.000     1.000
       0.000     0.000     1.000
       0.000     0.000     1.000
1
6         0.0       0.0      60.0
          0.0       0.0      90.0
          0.0       0.0     120.0
2         38        72
           2         2
```

436

```
        E5-DeP   nExSE ** Bypass but no blowdown or SE - sump water**
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1
     6     0.0     0.0    10.0
           0.0     0.0    15.0
           0.0     0.0    25.0
       Otherwise       ** Bypass and blowdown or SE --> CF **
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1
     6     0.0     0.0    30.0
           0.0     0.0    45.0                         ** blowdown **
           0.0     0.0    70.0
76 Does pressurization fail containment at VB?
     2      IP     nIP
     6       1       2
     3
     6      50      47      45      69      19      54
             2       1       1      -4  (  -10   +  -1)
       nE4-CFP E4nVent  E4nHWF VBreach  nTC-CV nE4-DHF
     5       1       2       4       5       6
        Pbase1   DP-H2  DPBlcw   DPWat   DPfVS
           AND
        NORMAL       2 !50 1 1 !50 1 2
                       !50 1 1 !50 1 2
                       !50 1 1 !50 1 2
                CFailP   Mean    Sigma
     7      69      19      54      53      51      51      75
             4  + ( 10       1) + (   3  * (  2  + (  3   *  -2)))
       nBreach   TC-CV  E4-DHF  E4-CL3  E4-CLD  E4-CLW    Vclr
     1       1
        PBase1
           AND
        THRESH       1   999.0
                         999.0
                         999.0
               Prevent containment failure
       Otherwise
     4       3       4       5       6
        Pbase2  DPBlow   DPWat   DPfVS
           AND
        NORMAL       2 !50 1 1 !50 1 2
                       !50 1 1 !50 1 2
                       !50 1 1 !50 1 2
                CFailP   Mean    Sigma
77 What is the base pressure before vessel breach?
     8   PB130   PB110    PB90    PB70    PB50    PB40    PB30    PB20
     6       1       2       3       4       5       6       7       8
     3
     4      50      47      45      69
             2       1       1      -4
```

```
      nE4-CFP E4nVent  E4nHWF VBreach
    2       1       2
        Pbase1   DP-H2
           AND
      GETHRESH       7   130.0   110.0    90.0    70.0    50.0    40.0
                         130.0   110.0    90.0    70.0    50.0    40.0
                         130.0   110.0    90.0    70.0    50.0    40.0
                       BASE PRESSURE BEFORE VESSEL BREACH
    2      69      53
            4   +   3
      nBreach  E4-CL3
    1       1
        Pbase1
           AND
      GETHRESH       7   999.0   999.0   999.0   999.0   999.0   999.0
                         999.0   999.0   999.0   999.0   999.0   999.0
                         999.0   999.0   999.0   999.0   999.0   999.0
                       CONTAINMENT HAS FAILED => BASE PRESSURE = 20PSI
      Otherwise
    1       3
        Pbase2
           AND
      GETHRESH       7   130.0   110.0    90.0    70.0    50.0    40.0
                         130.0   110.0    90.0    70.0    50.0    40.0
                         130.0   110.0    90.0    70.0    50.0    40.0
                       BASE PRESSURE BEFORE VESSEL BREACH
78 What is the pressure rise at vessel breach?
    8   DP130   DP110    DP90    DP70    DP50    DP40    DP30    DP20
    6       1       2       3       4       5       6       7       8
    3
    4      50      47      45      69
            2       1       1      -4
      nE4-CFP E4nVent  E4nHWF VBreach
    3       4       5       6
       DPBlow   DPWat   DPfVS
           AND
      GETHRESH       7   130.0   110.0    90.0    70.0    50.0    40.0
                         130.0   110.0    90.0    70.0    50.0    40.0
                         130.0   110.0    90.0    70.0    50.0    40.0
                       PRESSURE RISE AT VESSEL BREACH
    5      69      53      51      51      75
            4  + (  3  * (  2  + (  3   *  -2)))
      nBreach  E4-CL3  E4-CLD  E4-CLW    Vclr
    1       1
        Pbase1
           AND
      GETHRESH       7   999.0   999.0   999.0   999.0   999.0   999.0
                         999.0   999.0   999.0   999.0   999.0   999.0
                         999.0   999.0   999.0   999.0   999.0   999.0
                       CONTAINMENT HAS FAILED => DELTA P <30PSI
      Otherwise
    3       4       5       6
       DPBlow   DPWat   DPfVS
           AND
       GETHRESH      7   130.0   110.0    90.0    70.0    50.0    40.0
                         130.0   110.0    90.0    70.0    50.0    40.0
                         130.0   110.0    90.0    70.0    50.0    40.0
                       PRESSURE RISE AT VESSEL BREACH
79 Does direct melt-structure attack fail containment at VB?
     2      IM     nIM
     2       1       2
     7
     1      69
             4
       nBreach
         0.000   1.000
         0.000   1.000
         0.000   1.000
     2      69      70
             3       1
         Mthru nEBlowD
         0.100   0.900
         0.950   0.050
         1.000   0.000
     2      69      68
             3       1
         Mthru   Slump                 ** no dispersal **
         0.100   0.900
         0.900   0.100
         1.000   0.000
     1      69
             2
         SE-BH                         ** SE-BH **
       !79 2 1 !79 2 2
       !79 2 1 !79 2 2
       !79 2 1 !79 2 2
     3      69      68      74
             3       2       1
         Mthru    Flow   RCWet
         0.000   1.000
         0.400   0.600
         1.000   0.000
     2      69      68
             3       2
         Mthru    Flow
         0.000   1.000
         0.500   0.500
         1.000   0.000
       Otherwise       ** Alpha
         0.000   1.000
         0.000   1.000
         0.000   1.000
80 What is the location of containment failure after VB?
     4    nICL    ICLD    ICLW   ICLDW
     2       1       2       3       4
     5
     6      69      79      76      51      75      76
             1  + (  1       2)  +   2  + (  2       1)
```

439

```
         Alpha      IM     nIP  E4-CLD   nVclr      IP
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
         0.000   1.000   0.000   0.000
     2      79      76
             1       1
            IM      IP
       !28 4 1 !28 4 2   0.000 !28 4 3
       !28 4 1 !28 4 2   0.000 !28 4 3
       !28 4 1 !28 4 2   0.000 !28 4 3
     1      51
             3
        E4-CLW
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
         0.000   0.000   1.000   0.000
     1      76
             1
            IP
       !28 4 1 !28 4 2 !28 4 3   0.000
       !28 4 1 !28 4 2 !28 4 3   0.000
       !28 4 1 !28 4 2 !28 4 3   0.000
       Otherwise
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
         1.000   0.000   0.000   0.000
81 For drywell failures, is the failure at the head seal?
     3    IDHF    IDKF    nIDF
     2       1       2       3
     5
     1      79
             1
            IM
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     1      52
             1
        E4-DHF
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     1      52
             2
        E4-DKF
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     1      80
             2
          ICLD
       !29 3 1 !29 3 2 !29 3 3
       !29 3 1 !29 3 2 !29 3 3
       !29 3 1 !29 3 2 !29 3 3
       Otherwise
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
82 What is the containment leakage level after VB?
     3    nICL    ICL2    ICL3
     2       1       2       3
     5
     5      69      79      53      75      76
             1   +   1   +   3   +   2       1
         Alpha      IM  E4-CL3   nVclr      IP
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     3      80      81      76
         (   3    +  2)      1
          ICLW    IDKF      IP
       !30 6 1 !30 6 2 !30 6 3
       !30 6 1 !30 6 2 !30 6 3
       !30 6 1 !30 6 2 !30 6 3
     1      81
             1
          IDHF
       !30 7 1 !30 7 2 !30 7 3
       !30 7 1 !30 7 2 !30 7 3
       !30 7 1 !30 7 2 !30 7 3
     1      53
             2
        E4-CL2
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
83 Is the suppression pool drained following vessel breach?
     3   I-SPD   IpSPD   InSPD
     2       1       2       3
     4
     1      57
             1
        E5-SPD
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     1      57
             2
        E5pSPD
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     3      76      80      80
             1   (   3   +   4)
```

441

```
         IP     ICLW    ICLDW
      !31 1 1 !31 1 2 !31 1 3
      !31 1 1 !31 1 2 !31 1 3
      !31 1 1 !31 1 2 !31 1 3
      Otherwise
        0.000   0.000   1.000
        0.000   0.000   1.000
        0.000   0.000   1.000
84 Is ac power not available late?
     2     LfAC     L-AC
     2        1        2
     5
     3       32       19       24
              1        6        1
        E4fAC       TB      ReP @ 12.6 hours given none @ 11.6
        0.751    0.249
        0.925    0.075
        1.000    0.000
     2       32       19
              1        6
        E4fAC       TB
        0.683    0.317
        0.841    0.159
        0.924    0.076              TB but batteries survive
     2       32        6
              1        2
        E4fAC    E1-DC            @ 2.8 hours given none at 1.8
        0.614    0.386
        0.764    0.236
        0.886    0.114
     2       32        6
              1        1
        E4fAC    E1fDC            @ 2.8 hours given none at 1.8
        0.712    0.288
        0.886    0.114
        1.000    0.000
      Otherwise
        0.000    1.000
        0.000    1.000
        0.000    1.000
85 Is RHR not operating late?
     2    nLRHR     LRHR
     2        1        2
     3
     6       12       83       84       82       37       69
              1   +    1   +    1   + ( -1        1       -4)
       E1fRHR    I-SPD     LfAC      ICL    SatSP   Breach
        1.000    0.000
        1.000    0.000
        1.000    0.000
     2       12       32
              2        1
       E1rRHR    E4fAC
        0.001    0.999
```

```
         0.010   0.990
         0.500   0.500
       Otherwise
         0.000   1.000
         0.000   1.000
         0.000   1.000
86 Do containment sprays not operate following vessel breach?
     2    nLCS     LCS
     2       1       2
     3
     5      22      84      82      37      83
             1   +   1  + ( -1      1)   +   1
         E3fCS    LfAC     ICL   SatSP   I-SPD
         1.000   0.000
         1.000   0.000
         1.000   0.000
     1      40
             3
         E4-CS
         0.000   1.000
         0.000   1.000
         0.000   1.000
       Otherwise   (no prior spray)
         0.001   0.999
         0.010   0.990
         0.100   0.900
87 Is service water sprayed following vessel breach?
     2   nLSWS    LSWS
     2       1       2
     2
     4      22      55      84      86
             1   +   4   +   1   +   2
         E3fCS E5-HPSW    LfAC     LCS
         1.000   0.000
         1.000   0.000
         1.000   0.000
       Otherwise
         0.100   0.900
         0.500   0.500
         0.000   1.000
88 Is water not supplied to the debris late?
     2 nLDBWat  LDBWat
     2       2
     8
     1      84
             1
          LfAC
         1.000   0.000
         1.000   0.000
         1.000   0.000
     6      69      67      82      37      83      55
             4  + (  1  * (( 1   +   2     -1)   +   4))
       nBreach E5-LPIn    IncL  SubcSP nE5-SPD E5-HPSW
         0.000   1.000
         0.000   1.000
         0.000   1.000
     6      86      87      21      82      37      83
             2   +   2  + (  2   (   1   +   2)     -1)
           LCS    LSWS  E2rLPC    InCL  SubcSP nE5-SPD
         0.000   1.000
         0.000   1.000
         0.000   1.000
     5      67      21      37      53      57
             1   (   1 + (   1      -1   +   1))
       E5-LPIn  E2fLPC   SatSP   E5-CL  E5-SPD
         0.000   1.000
         0.000   1.000
         0.000   1.000
     3      13      32      38
            -1       1       1
       nE1fCOND  E4fAC  E5nDeP
         0.001   0.999
         0.010   0.990
         0.100   0.900
     1      67
             2
       E5-HPIn
         0.010   0.990
         0.100   0.900
         0.500   0.500
     3      54      55      55
             2   +   2   +   3
        E5rCRD E5rHPSW E5aHPSW
         0.010   0.990
         0.100   0.900
         0.500   0.500
       Otherwise
         1.000   0.000
         1.000   0.000
         1.000   0.000
89 Is the debris not coolable without water?
     2 nDBCwoW  DBCwoW
     2       1       2
     2
     3      69      71      72
         (   2   +   1   +   1)
         SE-BH      DH    ExSE *** dispersal ***
         0.900   0.100
         0.999   0.001
         1.000   0.000
       Otherwise
         1.000   0.000
         1.000   0.000
         1.000   0.000
90 Is the debris coolable with water?
     2    nDBC   DBCwW
     2       1       2
     5
     2       89      69
              2   +   4
         DBCwoW nBreach
          0.000   1.000
          0.000   1.000
          0.000   1.000
     3       69      71      72
              2   +   1   +   1
          SE-BH      DH    ExSE *** dispersal ***
          0.100   0.900
          0.500   0.500
          0.900   0.100
     1       68
              1
          Slump                 *** IDCOR says likely ***
          0.100   0.900
          0.900   0.100
          1.000   0.000
     5       67      82      37      83      55
              1  * (( 1   +   2     -1)   +  4)
       E5-LPIn     InCL  SubcSP nE5-SPD E5-HPSW
          0.100   0.900
          0.700   0.300
          0.900   0.100
       Otherwise                *** trickle out of vessel ***
          0.000   1.000
          0.500   0.500
          0.900   0.100
91 Is the debris not cooled?
     2 nCoolDB   CoolDB
     2        1       2
     3
     2       69      89
              4   +   2
       nBreach   DBCwoW
          0.000   1.000
          0.000   1.000
          0.000   1.000
     2       90      88
              2       2
          DBCwW  LDBWat
          0.000   1.000
          0.000   1.000
          0.000   1.000
       Otherwise
          1.000   0.000
          1.000   0.000
          1.000   0.000
92 Is the containment not vented late?
     2   LnVent  L-Vent
     2        1       2
     5
     2       82      69
              3   +   4
         ICL3 nBreach
        1.000   0.000
        1.000   0.000
        1.000   0.000
    2      84      46
            1       2
        E4fAC  E4nVth
      !47 7 1 !47 7 2
      !47 7 1 !47 7 2
      !47 7 1 !47 7 2
    4      91      85      38      48
            2       2       2       2
       CoolDB    LRHR  E5-Dep  E4nReP
        1.000   0.000
        1.000   0.000
        1.000   0.000
    3      84      32      46
            2   (   1   +   2)
        E4-AC   E4fAC  E4nVth
      !47 5 1 !47 5 2
      !47 5 1 !47 5 2
      !47 5 1 !47 5 2
      Otherwise
        1.000   0.000
        1.000   0.000
        1.000   0.000
93 Does the containment fail late due to temperature?
    2    LTCF   nLTCF
    2       1       2
    4
    4      69      80      80      88
            4   +   2   +   4   +   2
      nBreach    ICLD   ICLDW  LDBWat
        0.000   1.000
        0.000   1.000
        0.000   1.000
    1      89
            2
       DBCwoW
        1.000   0.000
        1.000   0.000
        1.000   0.000
    2      82      92
        (   3   +   2)
         ICL3  L-Vent          ** Reduced driving force
        0.010   0.990
        0.100   0.900
        0.500   0.500
      Otherwise
        0.100   0.900
        0.500   0.500
        0.900   0.100
94 Does the containment fail late?
    3    LPCF    LTCF    nLCF
```

```
     2        1       2       3
     7
     1       93
              1
           LTCF
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
     2       91      85
              2       2
         CoolDB    LRHR
          0.000   0.000   1.000
          0.000   0.000   1.000
          0.000   0.000   1.000
     6       91      85      82      92      88      69
              2       1       1       1       2      -4
         CoolDB   nLRHR    nICL  LnVent  LDBWat  Breach
          1.000   0.000   0.000
          1.000   0.000   0.000
          1.000   0.000   0.000
     4       91      85      82      92
              1       2       1       1
        nCoolDB    LRHR    nICL  LnVent ** non-condensibles **
          1.000   0.000   0.000
          1.000   0.000   0.000
          1.000   0.000   0.000
     4       91      82      92      88
              1       1       1       2
        nCoolDB    nICL  LnVent  LDBWat
          1.000   0.000   0.000
          1.000   0.000   0.000
          1.000   0.000   0.000
     3       91      82      92
              1       1       1
        nCoolDB    nICL  LnVent ** dh & non-condensibles **
          1.000   0.000   0.000
          1.000   0.000   0.000
          1.000   0.000   0.000
        Otherwise
          0.000   0.000   1.000
          0.000   0.000   1.000
          0.000   0.000   1.000
95 What is the location of late containment leakage?
     3     nLCL    LCLD    LCLW
     2        1       2       3
     6
     4       80      80      82      92
         (   2   +   4)  (   3   +   1)
           ICLD   ICLDW    ICL3  LnVent
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
     5       93      80      82      92      58
             1  ((( -3   +  -3)      1)  +  -3)
          LTCF   nICFW   nICL3  LnVent nE5-SPB3
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     2      80      92
             3   +   2
          ICLW  L-Vent
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     2      94      88
             1       2
          LPCF  LDBWat
       !28 4 1 !28 4 2 !28 4 3
       !28 4 1 !28 4 2 !28 4 3
       !28 4 1 !28 4 2 !28 4 3
     1      94
             1
          LPCF
         0.000   0.600   0.400
         0.000   0.960   0.040
         0.000   1.000   0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
96 For drywell failures, is the failure at the head seal?
     3    LDHF    LDKF    nLDF
     2       1       2       3
     6
     1      94
             2
          LTCF
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     1      81
             1
          IDHF
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     1      81
             2
          IDKF
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     2      94      88
             1       2
          LPCF  LDBWat
       !29 3 1 !29 3 2 !29 3 3
       !29 3 1 !29 3 2 !29 3 3
       !29 3 1 !29 3 2 !29 3 3
     1      95
             2
          LCLD
         0.100   0.900   0.000
         0.630   0.370   0.000
         0.900   0.100   0.000
       Otherwise
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
97 What is the level of late containment leakage?
     3    nLCL    LCL2    LCL3
     2       1       2       3
     8
     1      82
             3
          ICL3
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1      94
             2
          LTCF
         0.000   0.999   0.001
         0.000   0.500   0.500
         0.000   0.000   1.000
     2      94      96
             1       1
          LPCF    LDHF
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     3      94      88      95
             1   (   2   +   3)
          LPCF  LDBWat    LCLW
         0.000   0.900   0.100
         0.000   0.500   0.500
         0.000   0.100   0.900
     2      94      96
             1       2
          LPCF    LDKF
       !97 4 1 !97 4 2 !97 4 3
       !97 4 1 !97 4 2 !97 4 3
       !97 4 1 !97 4 2 !97 4 3
     1      92
             2
        L-Vent
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     1      82
             2
          ICL2
         0.000   1.000   0.000
```

449

```
         0.000   1.000   0.000
         0.000   1.000   0.000
       Otherwise
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
98 Is the suppression pool drained late?
     3    LSPD   pLSPD   nLSPD
     2       1       2       3
     4
     1      83
             1
         I-SPD
         1.000   0.000   0.000
         1.000   0.000   0.000
         1.000   0.000   0.000
     1      83
             2
         IpSPD
         0.000   1.000   0.000
         0.000   1.000   0.000
         0.000   1.000   0.000
     3      94      95      97
             1       3       3
          LPCF    LCLW    LCL3
       !31 1 1 !31 1 2 !31 1 3
       !31 1 1 !31 1 2 !31 1 3
       !31 1 1 !31 1 2 !31 1 3
       Otherwise
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
99 What is the level of late suppression pool bypass?
     3   nLSPB   LSPB2   LSPB3
     2       1       2       3
     6
     5      98      95      98      58      69
             1 + ((  2  +    2   +  3)     -4)
          LSPD    LCLD   pLSPD E5-SPB3  Breach
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     3      20      26      49
         (   1   +   1   +   1)
        PCycBk   RePBk E4-RePB
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
     5      69      95      41      38      41
             4       2 (     3  +(   1       2))
       nBreach    LCLD E4-FLD3  E4nDeP E4-FLD2
         0.000   0.000   1.000
         0.000   0.000   1.000
         0.000   0.000   1.000
```

```
      3      69      98      25
              4       2       1
        nBreach   pLSPD oSRVBkr
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
      3      69      95      25
              4       2       1
        nBreach    LCLD oSRVBkr
          0.000   1.000   0.000
          0.000   1.000   0.000
          0.000   1.000   0.000
        Otherwise
          1.000   0.000   0.000
          1.000   0.000   0.000
          1.000   0.000   0.000
100 Do drywell sprays continue?
      2   nVLDS    VLDS
      2       1       2
      4
      2      86      87
              1       1
           nLCS   nLSWS
          1.000   0.000
          1.000   0.000
          1.000   0.000
      6      86      94      37      98      55      55
              2   (  -3       1   +   1)   (  1   +   4)
            LCS     LCF   SatSP    LSPD E5fHPSW E5-HPSW
          1.000   0.000
          1.000   0.000
          1.000   0.000
      4      86      94      37      98
              2   (  -3       1   +   1)
            LCS     LCF   SatSP    LSPD
        !87 2 1 !87 2 2
        !87 2 1 !87 2 2
        !87 2 1 !87 2 2
        Otherwise
          0.000   1.000
          0.000   1.000
          0.000   1.000
101 What is the level of late RB bypass without a burn?
      4   nLRBB   LRBB2   LRBB3   LRBB4
      2       1       2       3       4
      8
      1      69
              1
          Alpha
          0.000   0.000   0.000   1.000
          0.000   0.000   0.000   1.000
          0.000   0.000   0.000   1.000
      6      79      92      84      59      59      63
          (   1  +  ( 2       2))  (  3   +   4   +  -4)
          IM  L-Vent    L-AC  E5-RBB3  E5-RBB4   nERRB4
       0.000   1.000   0.000   0.000
       0.000   1.000   0.000   0.000
       0.000   1.000   0.000   0.000
   2      63      92
           4   +   2
       ERRB4  L-Vent
       0.000   0.000   0.000   1.000
       0.000   0.000   0.000   1.000
       0.000   0.000   0.000   1.000
   1      63
           3
       ERRB3
       0.000   0.000   1.000   0.000
       0.000   0.000   1.000   0.000
       0.000   0.000   1.000   0.000
   1      96
           1
        LDHF
       0.000   0.000   1.000   0.000
       0.000   0.000   1.000   0.000
       0.000   0.000   1.000   0.000
   2      97      63
           3   +   2
        LCL3   ERRB2
       0.000   1.000   0.000   0.000
       0.000   1.000   0.000   0.000
       0.000   1.000   0.000   0.000
   1      97
           2
        LCL2
       0.900   0.100   0.000   0.000
       0.500   0.500   0.000   0.000
       0.100   0.900   0.000   0.000
     Otherwise
       1.000   0.000   0.000   0.000
       1.000   0.000   0.000   0.000
       1.000   0.000   0.000   0.000
102 Are fire systems operating late without a late HB?
   2  LFireS nLFireS
   2       1       2
   6
   1      64
           1
      EFireS
       1.000   0.000
       1.000   0.000
       1.000   0.000
   1      97
           1
        nLCL
       0.000   1.000
       0.000   1.000
       0.000   1.000
       1      101
                4
            LRBB4
            0.000   1.000
            0.000   1.000
            0.000   1.000
       1       96
                1
             LDHF
            0.000   1.000
            0.000   1.000
            0.000   1.000
       3       97      99      37
                3       1       2
             LCLW   nLSPB  SubcSP
            0.000   1.000
            0.000   1.000
            0.000   1.000
        Otherwise
            1.000   0.000           *** LFireS ***
            1.000   0.000
            1.000   0.000
   103 Does standby gas treatment not work late without late HB?
      2   nLSGTS   LSGTS
      2        1       2
      2
      3       84     102      65
               2       2      -1
            L-AC nLFireS nEfSGTS
           0.000   1.000
           0.000   1.000
           0.000   1.000
       Otherwise
           1.000   0.000
           1.000   0.000
           1.000   0.000
   104 Does H2 burn in the reactor building after vessel breach?
      2      LHB    nLHB
      2        1       2
      5
      1      101
               4
           LRBB4
           0.000   1.000
           0.000   1.000
           0.000   1.000
      1       91
               2
          CoolDB
           0.000   1.000
           0.000   1.000
           0.000   1.000
      2       97     103
               2       2
          LCL2    LSGTS
         0.100    0.900
         0.500    0.500
         0.900    0.100
     2      97      103
            -1        1
           LCL  nLSGTS
         0.500    0.500
         1.000    0.000
         1.000    0.000
       Otherwise
         0.000    1.000
         0.000    1.000
         0.000    1.000
105 What is the level of late RB bypass?
     4   nLRBB    LRBB2    LRBB3    LRBB4
     2       1        2        3        4
     6
     1     101
             4
         LRBB4
         0.000    0.000    0.000    1.000
         0.000    0.000    0.000    1.000
         0.000    0.000    0.000    1.000
     2     104      101
             1        3
           LHB    LRBB3
       !63 2 1 !63 2 2 !63 2 3 !63 2 4
       !63 2 1 !63 2 2 !63 2 3 !63 2 4
       !63 2 1 !63 2 2 !63 2 3 !63 2 4
     3     104      101      101
             1    (   1    +   2)
           LHB    nLRBB    LRBB2
       !63 3 1 !63 3 2 !63 3 3 !63 3 4
       !63 3 1 !63 3 2 !63 3 3 !63 3 4
       !63 3 1 !63 3 2 !63 3 3 !63 3 4
     1     101
             3
         LRBB3
         0.000    0.000    1.000    0.000
         0.000    0.000    1.000    0.000
         0.000    0.000    1.000    0.000
     3     101       97      103
             2   + (  2        1)
         LRBB2     LCL2   nLSGTS
         0.000    1.000    0.000    0.000
         0.000    1.000    0.000    0.000
         0.000    1.000    0.000    0.000
       ,Otherwise
         1.000    0.000    0.000    0.000
         1.000    0.000    0.000    0.000
         1.000    0.000    0.000    0.000
106 Are fire systems operating late?
     2  LFireS nLFireS
```

```
      2       1       2
      3
      2     102     104
              2       2
        nLFireS    nLHB
          0.000   1.000
          0.000   1.000
          0.000   1.000
      3      96     102     104
              1       2       1
           LDHF nLFireS     LHB
          0.000   1.000
          0.000   1.000
          0.000   1.000
        Otherwise
          1.000   0.000
          1.000   0.000
          1.000   0.000
107 Does standby gas treatment work late?
      2  nLSGTS   LSGTS
      2       1       2
      2
      2     103     106
              2       2
          LSGTS nLFireS
          0.000   1.000
          0.000   1.000
          0.000   1.000
        Otherwise
          1.000   0.000
          1.000   0.000
          1.000   0.000
```

DISTRIBUTION:

U. S. Government Printing Office
Receiving Branch (Attn: NRC Stock)
8610 Cherry Lane
Laurel, MD 20707
100 copies for AN

3141	S. A. Landenberger (5)
3151	W. L. Garner
6400	D. J. McCloskey
6410	N. R. Ortiz
6414	A. S. Benjamin (25)
6412	A. L. Camp
6412	F. T. Harper
6412	D. M. Kunsman
6512	D. M. Ericson, Jr.
6415	F. E. Haskin
6415	J. L. Sprung
6513	D. D. Carlson
6420	J. V. Walker
6422	D. A. Powers
6427	M. Berman
6440	D. A. Dahlgren
6442	W. A. von Riesemann
6449	K. D. Bergeron
6449	D. C. Williams
8024	P. W. Dean

456

Frank Abbey
U.K. Atomic Energy Authority
Wigshaw Lane, Culcheth
Warrington, Chesire
United Kingdom, WA3 4NE

Steven Aftergood
Committee to Bridge the Gap
1637 Butler Avenue, #203
Los Angeles, CA 90025

C. N. Amos
Technadyne, Inc.
300 Virginia SE
Albuquerque, NM 87108

George Apostolakis
University of California, Los Angeles
Boelter Hall, Room 5532
Los Angeles, CA 90024

James W. Ashkar
Boston Edison Company
800 Boylston Street
Boston, MA 02199

Donald H. Ashton
Bechtel Power Corporation
15740 Shady Grove Road
Gaithersburg, MD 20877

Mark Averett
Florida Power Corporation
P.O. Box 14042
St. Petersburg, FL 33733

Michael B. Aycock
TERA Corporation
7101 Wisconsin Avenue
Bethesda, MD 20814

Raymond O. Bagley
Northeast Utilities
P.O. Box 270
Hartford, CT 06101

J. Bagues
Consejo de Seguridad Nucleare
Sorangela de la Cruz 3
Madrid 38046
Spain

457

H. Bairiot
Belgonucleaire S A
Rue de Champ de Mars 25
B-1050 Brussels
Belgium

Fred J. Balkovetz
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

Patrick Baranowsky
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

Robert A. Bari
Brookhaven National Laboratories
Building 130
Upton, NY 11973

Richard Barrett Baskin
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

S. Kenneth
S. California Edison Company
P.O. Box 800
Rosemead, CA 91770

Werner Bastl
Gesellschaft Fur Reaktorsicherheit
Forschungsgelande
8046 Garching
Federal Republic of Germany

Paul Baybutt
Battelle Columbus Division
505 King Avenue
Columbus OH 43201

Bruce B. Beckley
Public Service Company
P.O. Box 330
Manchester, NH 03105

Myer Bender
Querytech, Inc.
408 Cedar Bluff Road
Knoxville, TN 37923

R. C. Bertucio
Energy Incorporated
1851 So. Central Place, Suite 201
Kent, WA 98031

Peter Bieniarz
Risk Management Association
2309 Dietz Farm Road, NW
Albuquerque, NM 87107

Adolf Birkhofer
Gesellschaft Fur Reaktorsicherheit
Forschungsgelande
8046 Garching
Federal Republic of Germany

James Blackburn
Illinois Dept. of Nuclear Safety
1035 Outer Park Drive
Springfield, IL 62704

Dennis C. Bley
Pickard, Lowe & Garrick, Inc.
2260 University Drive
Newport Beach, CA 92660

Roger M. Blond
Science Applications Int. Corp.
1710 Goodridge Drive
McLean, VA 22102

A. John Bourne
U.K. Atomic Energy Authority
Wigshaw Lane, Culcheth
Warrington, Chesire
United Kingdom, WA3 4NE

G. J. Boyd
Safety & Reliability Optimization
Services
9724 Kingston Pike
Knoxville, TN 37922

K. J. Brinkmann
Netherlands Energy Res. Fdtn.
P.O. Box 1
1755ZG Petten NH
Netherlands

459

R. H. Buchholz
Nutech
6835 Via Del Oro
San Jose, CA 95119

Fred Buckman
Delian Corporation
296 Duff Road
Monroeville, PA 15146

Robert J. Budnitz
Future Resources Associates
734 Alameda
Berkeley, CA 94707

A. R. Buhl
International Technology Corp.
575 Oak Ridge Turnpike
Oak Ridge, TN 37830

Gary R. Burdick
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

Kenneth S. Canady
Duke Power Company
422 S. Church Street
Charlotte, NC 28217

Lennart Carlsson
Swedish Nuclear Power Inspectorate
Box 27106
Stockholm, Sweden 08-635560
Sweden

Annick Carnino
Electricite de France
32 Rue de Monceau 8EME
Paris, France F5008
France

N. G. Cathey
Idaho National Engineering Lab.
P. O. Box 1625
Idaho Falls, ID 83415

Ivan Catton
22436 Galilee Street
Woodland Hills, CA 91364

460

S. Chakraborty
Radiation Protection Section
Div. De La Securite Des Inst. Nuc.
5303 Wurenlingen, Switzerland
Switzerland

Sen-I Chang
Institute of Nuclear Energy Research
P.O. Box 3
Lungtan, 325 Taiwan

Robert F. Christie
Tennessee Valley Authority
400 W. Summit Hill Avenue, W10D1
90 Knoxville, TN 37902

Frank Coffman
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

Larry L. Conradi
Energy Incorporated
1851 South Central Place, Suite 201
Kent, WA 98031

Peter Cooper
U.K. Atomic Energy Authority
Wigshaw Lane, Culcheth
Warrington, Chesire
United Kingdom, WA3 4NE

Michael Corradini
University of Wisconsin
1500 Johnson Drive
Madison, WI 53706

Michael Cullingford
IAEA
P.O. Box 200
A-1400 Vienna, Austria

Garth Cummings
Lawrence Livermore Laboratory
L-91, Box 808
Livermore, CA 94526

Mark A. Cunningham
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

461

James J. Curry
7135 Salem Park Circle
Mechanicsburg, PA 17055

Peter Cybulskis
Battelle Columbus Division
505 King Avenue
Columbus, OH 43201

Peter R. Davis
Intermountain Technologies, Inc.
P.O. Box 1604
Idaho Falls, ID 83401

Jose DeCarlos
Sor Angela De la Cruz, No. 3, Planta 8
Madrid 28016, Spain

Richard Deem
New York Power Authority
123 Main Street
White Plains, NY 10601

Richard S. Denning
Battelle Columbus Division
505 King Avenue
Columbus, OH 43201

Raymond DiSalvo
Battelle Columbus Division
505 King Avenue
Columbus, OH 43201

Adel A. El-Bassioni
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

J. Mark Elliott
International Energy Associates, Ltd.
600 New Hampshire Ave. NW, Suite 600
Washington, DC 20037

Energy Issues Rep.
Sierra Club
1228 North Street#31
Sacramento, CA 95814

Malcolm L. Ernst
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, GA 30323

Executive Director
Natural Resources Defense Council
122 East 42nd Street
New YorkNY 10168

P. Fehrenback
Atomic Energy Canada, Ltd.
Chalk River
Ontario, Canada

A. Fiege
Kernforschungszentrum
Postfach 3640
7500 Karlsruhe
Federal Republic of Germany

George F. Flanagan
Oak Ridge National Laboratories
P.O. Box Y
Oak Ridge, TN 37831

Mario Fontana
International Technology Corp.
575 Oak Ridge Turnpike
Oak Ridge, TN 37830

Ray R. Fray
Pacific Gas and Electric Co.
215 Market Street, Room 206
San Francisco, CA 94106

James Fulford
NUS Corporation
910 Clopper Road
Gaithersburg, MD 20878

J. B. Fussell
JBF Associates, Inc.
1630 Downtown West Boulevard
Knoxville, TN 37919

Raymond H. V. Gallucci
Battelle Pacific Northwest Labs.
P.O. Box 999
Richland, WA 99352

John Garrick
Pickard, Lowe & Garrick, Inc.
2260 University Drive
Newport Beach, CA 92660

John Gaunt
British Embassy
3100 Massachusetts Avenue, NW
Washington, DC 20008

J. Gieseke
Battelle Columbus Division
505 King Avenue
Columbus, OH 43201

James C. Glynn
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

L. Hammar
Statens Karnkraftinspektion
P.O. Box 27106
S-10252 Stockholm, Sweden

M. R. Hayns
U.K. Atomic Energy Authority
Wigshaw Lane, Culcheth
Warrington, Chesire
United Kingdom WA3 4NE

Robert E. Henry
Fauske and Associates, Inc.
16W070 West 83rd Street
Burr Ridge, IL 60521

Daniel Hirsch
University of California
A. Stevenson Prog. on Nuclear Policy
Santa Cruz, CA 95064

Steven Hodge
Oak Ridge National Laboratories
P.O. Box Y
Oak Ridge, TN 37831

Thomas C. Houghton
KMC, Inc.
1747 Pennsylvania Avenue, NW
Washington, DC 20006

Dean Houston
U.S. Nuclear Regulatory Commission
Advisory Comm. on Reactor Safeguards
Washington, DC 20555

Der Yu Hsia
Taiwan Atomic Energy Council
67, Lane 144, Keelung Rd., Sec. 4
Taipei, Taiwan

R. E. Jaquith
Combustion Engineering, Inc.
1000 Prospect Hill Road
M/C 9490-2405
Windsor, CT 06095

Kjell Johannson
Studsvik Energiteknik AB
S-611 82, Nykoping, Sweden

D. H. Johnson
Pickard, Lowe & Garrick, Inc.
2260 University Drive
Newport Beach, CA 92660

Vojin Joksimovich
NUS Corporation
16885 W. Bernardo Drive
San Diego, CA 92127

H. R. Jun
Korea Adv. Energy Research Inst.
P.O. Box 7, Daeduk Danju
Chungnam 300-31, Korea

Peter Kafka
Gesellschaft Fur Reaktorsicherheit
Forschungsgelande 8046 Garching
Federal Republic of Germany

M. Kahatib-Rahbar
Brookhaven National Laboratory
Associated Universities, Inc.
Upton, NY 11973

Geoffrey D. Kaiser
NUS Corporation
910 Clopper Road
Gaithersburg, MD 20878

W. Kastenburg
UCLA
5532 Boelter Hall
Los Angeles, CA 90024

465

W. Kato
Brookhaven National Laboratory
Associated Universities, Inc.
Upton, NY 11973

William Kerr
University of Michigan, North Campus
Department of Nuclear Engineering
Ann Arbor, MI 48109

George Klopp
Commonwealth Edison Company
P.O. Box 767, Room 35W
Chicago, IL 60690

Klaus Koberlein
Gesellschaft Fur Reaktorsicherheit
Forschungsgelande 8046 Garching
Federal Republic of Germany

E. Kohn
Atomic Energy Canada Ltd.
Candu Operations, Mississauga
Ontario, Canada

Herbert J. C. Kouts
Brookhaven National Laboratories
Building 179C
Upton, L.I. NY 11973

Thomas Kress
Oak Ridge National Laboratories
P.O. Box Y
Oak Ridge, TN 37831

Jeanne-Marie Lanore
CEA B.P. No. 6
92660 Fonenay-Au-Roses, France

John C. Lee
University of Michigan, North Campus,
Dept. of Nuclear Engineering
Ann Arbor, MI 48109

Harold W. Lewis
University of California
Physics Department
Santa Barbara, CA 93106

Librarian
Atomic Industrial Forum
7101 Wisconsin Avenue
Bethesda, MD 20814

Eng Lin
Taiwan Power Company
242, Roosevelt Rd., Sec. 3
Taipei, Taiwan

J. P. Longworth
Central Electric Gen'Ting Brd.
Berkeley Gloucester
GL13 9PB United Kingdom

William J. Luckas
Brookhaven National Laboratories
Building 130
Upton, L.I., NY 11973

Hans Ludewig
Brookhaven National Laboratories
Building 130
Upton, L.I., NY 11973

P. MacDonald
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

A. P. Malinauskas
Oak Ridge National Laboratories
P.O. Box Y
Oak Ridge, TN 37831

Giuseppe Mancini
Commission European Comm.
CEC-JRC Eraton
Ispra Varese, Italy

Alan McGowan
Scientists Institute for Public Info.
355 Lexington Avenue
New York, NY 10017

Ralph Meyer
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

Joselyn Mitchell
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

Joseph A. Murphy
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

Sue Niemczyk
1545 18th St. NW, #112
Washington, DC 20036

Edward P. O'Donnell
Ebasco Services, Inc.
2 World Trade Center, 89th Floor
New York, NY 10048

David Okrent
University of California, Los Angeles
Boelter Hall, Room 5532
Los Angeles, CA 90024

William Parkinson
Delian Corporation
1340 Saratoga-Sunnyvale Road
Suite 206
San Jose, CA 95129

G. Petrangeli
ENEA Nucl. Energ. ALT Disp
Via V Brancati 48
00144 Roma, Italy

Robert D. Pollard
Union of Concerned Scientists
1616 P Street, NW, Suite 310
Washington, DC 20036

R. Potter
UK Atomic Energy Authority
Winfrith Dorchester Dorset
DT2 8DH United Kingdom

William T. Pratt
Brookhaven National Laboratories
Building 130
Upton, NY 11973

David Pyatt
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

Ingvard Rasmussen
Riso National Laboratory
Postbox 49 DK-4000
Roskilde, Denmark

Norman C. Rasmussen
Massachusetts Institute of Technology
77 Massachusetts Avenue
Cambridge, MA 02139

Dennis Richardson
Westinghouse Electric Corporation
P. O. Box 355
Pittsburg, PA 15230

Robert Ritzman
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94304

Denwood F. Ross
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

Frank Rowsome
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

John Siegal
Atomic Industrial Forum
7101 Wisconsin Avenue
Bethesda, MD 20814

Chester P. Siess
Advisory Comm. on Reactor Safeguards
805 Hamilton Drive
Champaign, IL 61820

Melvin Silberberg
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

469

K. Soda
Japan Atomic Energy Res. Inst.
Tokai-Mura Naka-Gun
IBARAKI-KEN 319-11, Japan

Leonard Soffer
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

Themis P. Speis
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

Klaus B. Stadie
OECD-NEA, 38 Bld. Suchet
75016 Paris, France

Wayne L. Stiede
Commonwealth Edison Company
P.O. Box 767
Chicago, IL 60690

John Taylor
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94303

Merrill A. Taylor
U.S. Nuclear Regulatory Commission
Office of the EDO
Washington, DC 20555

Technical Library
Electric Power Research Institute
P.O. Box 10412
Palo AltoCA 94304

Mark I. Temme
General Electric, Inc.
P.O. Box 3508
Sunnyvale, CA 94088

T. G. Theofanous
University of California, S.B.
Dept. of Chemical and Nuclear Eng.
Santa Barbara, CA 93106

470

BEST COPY AVAILABLE

Gordon Thompson
Inst. for Resource & Security
  Studies
27 Elsworth Avenue
Cambridge, MA 02139

Long Sun Tong
9733 Lookout Place
Gaithersburg, MD 20879

A. F. Torri
Pickard, Lowe & Garrick, Inc.
191 Calle Magdalena, Suite 290
Encinitas, CA 92024

Chao-Chin Tung
Atomic Energy Council
67 Lane 144, Sec 4, Keelung Rd.
Taipei, Taiwan

Ian B. Wall
Electric Power Research Institute
P.O. Box 10412
Palo Alto, CA 94303

David Ward
Savannah River Laboratories
Building 773-A, A-219
Aiken, SC 29808

Edward Warman
Stone & Webster Engineering Corp.
P.O. Box 2325
Boston, MA 02107

Norman Weber
Sargent & Lundy Co.
55 E. Monroe Street
Chicago, IL 60603

Lois Webster
American Nuclear Society
555 N. Kensington Avenue
La Grange Park, IL 60525

Paul Wood
Delian Corporation
296 Duff Road
Monroeville, PA 15146

John Wreathall
Battelle Columbus Division
505 King Avenue
Columbus, OH 43201

Roger Wyrick
Inst. for Nuclear Power Operations
1100 Circle 75 Parkway, Suite 1500
Atlanta, GA 30339

Jonathan Young
Energy, Inc.
1851 South Central Place
Suite 201
Kent, WA 98031

Steven Sholly
MAB Technical Associates
1723 Hamilton Avenue, Suite K
San Jose, CA 95125

Sam Hobbs
Mississippi Power & Light
Post Office box 1640
Jackson, Mississippi 39215-1640

Gary Smith
Mississippi Power & Light
Post Office box 1640
Jackson, Mississippi 39215-1640

Mr. A. R. Diederich
Supervising Engineer
Environmental Branch
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Greg Krueger
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Margaret Federline
U.S. Nuclear Regulatory Commission
Office of the Commission
Washington, D.C. 20555

Walter Lowenstein
Electric Power Research Institute
Post Office Box 10412
Palo Alto, California 94303

William Vesely
Science Applications Int. Corp.
2929 Kenny Road, Suite 245
Columbus, Ohio 43221

Gareth Parry
NUS Corporation
910 Clopper Road
Gaithersburg, Maryland 20878

Doug Rickeard
Virginia Electric Power Company
James River Plaza
Post Office Box 26666
Richmond, Virginia 23261

R. J. Hardwick, Jr.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, Virginia 23060

Roberta Gallante
Tennessee Valley Authority
400 West Summit Hill Drive
W4A24C-k
Knoxville, Tennessee 37902

NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1 REPORT NUMBER *(Assigned by TIDC, add Vol. No., if any)*

NUREG/CR-4700
SAND86-1135, Vol. 3

2 TITLE AND SUBTITLE

CONTAINMENT EVENT ANALYSIS FOR POSTULATED SEVERE ACCIDENTS: PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

3 LEAVE BLANK

4 DATE REPORT COMPLETED

MONTH	YEAR
May	1987

5 AUTHOR(S)

C. N. Amos, J. M. Griesmeyer, A. M. Kolaczkowski

6 DATE REPORT ISSUED

MONTH	YEAR
May	1987

7 PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*

Sandia National Laboratories
Albuquerque, NM 87185

8 PROJECT/TASK/WORK UNIT NUMBER

9 FIN OR GRANT NUMBER

A1322

10 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*

Division of Reactor System Safety
Office of Nucler Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a TYPE OF REPORT

b PERIOD COVERED *(Inclusive dates)*

12 SUPPLEMENTARY NOTES

13 ABSTRACT *(200 words or less)*

A study has been performed as part of the Severe Accident Risk Reduction Program (SARRP) to investigate the response of a particular boiling water reactor with a Mark I containment (Peach Bottom Unit 2) to postulated severe accidents. A detailed containment event tree for the Peach Bottom plant has been developed to describe the various possible accident pathways that can lead to radioactive releases from containment. Data and analyses from a large number of NRC and industry-sponsored programs have been reviewed and used as a basis for quantifying the event tree, i.e., determining the likelihood of the pathways at each branch point for a variety of accident sequence initiators. A generalized containment event tree code, called EVNTRE, has been developed to facilitate the quantification. The uncertainty in the results has been examined by performing the quantification three times, using a different set of input each time to represent the variation of opinion in the reactor safety community. In the so-called "central" estimate, the likelihood of early containment failure (occurring before or within a short time after reactor vessel breach) was found to be significant because of the possible occurrence of the following phenomena that can threaten containment integrity: (1) meltthrough of the drywell shell caused by thermal attack from core debris, and (2) drywell overpressurization caused by rapid depressurization of the reactor vessel in combination with other events such as direct heating. However, uncertainties surrounding these issues could cause the early failure likelihood to be significantly lower than in the central estimate. This work supports NRC's assessment of severe accident risks to be published in NUREG-1150.

14 DOCUMENT ANALYSIS – a KEYWORDS/DESCRIPTORS

Severe Accident Risk Reduction Program (SARRP)
Peach Bottom Plant
Event Tree Code (EVNTRE)
Containment integrity

b IDENTIFIERS/OPEN-ENDED TERMS

15 AVAILABILITY STATEMENT

Unlimited

16 SECURITY CLASSIFICATION

(This page)

Unclassified

(This report)

17 NUMBER OF PAGES

18 PRICE

474

BEST COPY AVAILABLE